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

April 16, 2015

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 – NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000285/2015007

Dear Mr. Cortopassi:

On March 13, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. The NRC inspectors discussed the results of this inspection with yourself and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented ten findings of very low safety significance (Green) in this report. Nine of these findings involved violations of NRC requirements; one of these violations was determined to be a Severity Level IV under the traditional enforcement process; and one finding did not involve a violation of NRC requirements. Further, inspectors documented a licensee-identified violation which was determined to be of very low safety significance in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Fort Calhoun Station.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement 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 Fort Calhoun Station.

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

Sincerely,

/RA/

Thomas R. Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000285/2015007
w/Attachment: Supplemental Information

Electronic Distribution for Fort Calhoun Station

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	05000285
License:	DPR-40
Report No.:	05000285/2015007
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	February 9 through March 13, 2015
Team Leader:	J. Dixon, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	J. Bozga, Reactor Inspector, Engineering Branch 1, Region III J. Braisted, Reactor Inspector, Engineering Branch 1 S. Bussey, Reactor Technology Instructor, Technical Training Center B. Correll, Reactor Inspector, Engineering Branch 2 C. Smith, Reactor Inspector, Engineering Branch 1
Accompanying Personnel:	S. Gardner, Contractor, Beckman and Associates M. Yeminy, Contractor, Beckman and Associates
Approved By:	Thomas R. Farnholtz, Branch Chief Engineering Branch 1

SUMMARY

IR 05000285/2015007; 02/09/2015 – 03/13/2015; Fort Calhoun Station; Component Design Basis Inspection.

The inspection activities described in this report were performed between February 9, 2015, and March 13, 2015, by four inspectors from the NRC's Region IV office, one inspector from the NRC's Region III office, one instructor from the NRC's Technical Training Center, and two contractors. Ten findings of very low safety significance (Green) are documented in this report. Nine of these findings involved violations of NRC requirements; one of these violations was determined to be Severity Level IV under the traditional enforcement process; and one was a finding that did not involve a violation of NRC requirements. Additionally, a licensee-identified violation of very low safety significance is documented in this report. The significance of inspection findings is indicated by their color (Green, White, Yellow, or Red), which is determined using Inspection Manual Chapter 0609, "Significance Determination Process." Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects Within the Cross-Cutting Areas." Violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process."

Cornerstone: Mitigating Systems

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis…are correctly translated into specifications, drawings, procedures, and instructions." Specifically, prior to March 13, 2015, the licensee failed to ensure that battery sizing and load profile calculations included proper design data for inrush currents, a random load, and possible worst case load currents. In response to these issues, the licensee updated the design values to account for the missed loads to ensure the batteries maintained adequate available margin. This finding was entered into the licensee's corrective action program as Condition Report CR 2014-14857.

  The team determined that the failure to adequately perform a battery sizing and load profile calculation, to ensure proper battery size and margin was maintained, was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to account for inrush currents, random loads, and worst case load currents during load profile and battery sizing calculations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.1.b.1)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis…are correctly translated into specifications, drawings, procedures, and instructions." Specifically, since 2009, the licensee failed to update battery maintenance procedures with the current maximum intercell resistance values. In response to this issue, the licensee performed a visual inspection of the battery intercell connections, performed a review of the latest intercell resistance measurements to identify any values that exceeded the correct acceptance criteria value, and performed an immediate operability determination. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02129.

  The team determined that the failure to establish the correct acceptance criteria values for battery intercell resistance measurements was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee had incorrect acceptance criteria for maximum intercell connection resistance measurements, and failed to identify an intercell connection that should have been disassembled, cleaned, reassembled, and remeasured. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of human performance associated with documentation because the licensee failed to maintain complete, accurate, and up-to-date documentation. [H.7] (Section 1R21.2.1.b.2)

- Green. The team identified a Green finding for the licensee's failure to verify or check the adequacy of design of the 125 Vdc batteries from environmental effects. Specifically, the licensee failed to account for the effects of elevated battery room temperature on expected battery service life, in accordance with EPRI Standard TR-100248, "Stationary Battery Guide: Design Application, and Maintenance," Revision 2. In response to this issue, the licensee performed an immediate operability determination to evaluate the effects of the elevated battery room temperatures and to determine when to modify the testing frequency based on the shorter life of the batteries. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02390.

  The team determined that the failure to account for elevated battery room temperature effects on battery service life was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, if left uncorrected, it could lead to a more significant safety concern in that the batteries could fail to maintain sufficient capacity and go undetected when testing at the normal 5 year interval. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-

Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of problem identification and resolution associated with operating experience because the licensee failed to evaluate and implement the EPRI standard based on industry experience when measuring room temperature readings above the optimal battery room temperature. [P.5] (Section 1R21.2.1.b.3)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to March 12, 2015, the licensee failed to verify or check the adequacy of the reactor protective system power supplies: 1) service life as a function of expected life minus shelf life; 2) vendor requirements for in-storage and post-storage maintenance; and 3) including or addressing laboratory failure analysis conclusions that a required component was, although functional, at its "end of life" after 18 years. In response to this issue, the licensee performed an immediate operability determination, verified the power supply's ripple checks were within tolerance, performed an engineering evaluation to support an operable but non-conforming condition, and generated rework activities to replace/refurbish the installed power supplies. This finding was entered into the licensee's corrective action program as Condition Reports CR 2015-02809 and CR 2015-02811.

  The team determined that the failure to perform an adequate justification for having reactor protective system power supplies installed beyond vendor recommend life was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to perform an adequate justification for continued operation for reactor protective system power supplies that were beyond vendor recommended life. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.5)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to March 13, 2015, the licensee failed to

perform an adequate design review to upgrade the auxiliary building single failure proof crane capacity, by failing to comply with ASME NOG-1-2004, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)." In response to this issue, the licensee performed an operability determination and concluded that the crane was operable but non-conforming, and limited the use of the main hook to the original 75 ton value until the long term actions can be completed to restore the crane to fully operable. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02718.

The team determined that the failure to perform an adequate design review to upgrade the auxiliary building single failure proof crane capacity was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, the licensee failed to comply with ASME NOG-1-2004 requirements to ensure the auxiliary building crane remained elastic when subjected to design loads for safe load handling of heavy loads. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened to Exhibit 4, "External Events Screening Questions," because it was a function specifically designed to mitigate a seismic event. Per Exhibit 4 the issue screened to a more detailed risk evaluation because if the seismic function were assumed to be completely failed and a load were dropped it would impact the spent fuel pool cooling or the safety injection refueling water storage tank functions. Therefore, the Region IV senior reactor analyst performed a more detailed risk evaluation. Given that the frequency of the initiating event is less than 1×10^{-6}, the analyst determined that the finding was of very low safety significance (Green). The team determined that this finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.8)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to March 13, 2015, the licensee did not verify the adequacy of the design calculation or a suitable testing program to ensure the required net positive suction head was available for the turbine-driven auxiliary feedwater pump. In response to this issue, the licensee performed an operability determination; revised several calculational errors, including removing conservatisms which resulted in a gain of net positive suction head; and contacted the original equipment manufacturer who provided a testing summary that determined the turbine-driven pump could operate for a period of time below the required net positive suction head. This provided the licensee with the basis for an operable but non-conforming condition. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02414.

  The team determined that the failure to verify the adequacy of the auxiliary feedwater system design through calculational analysis and a suitable test program was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability, and capability of systems that

respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to have adequate measures in place to ensure an acceptable design analysis and a suitable test program to verify the design inputs and ensure the capability of the auxiliary feedwater system to perform its safety function. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of human performance associated with conservative bias because individuals failed to use decision making practices that emphasize prudent choices over those that are simply allowed. [H.14] (Section 1R21.2.9)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to March 13, 2015, the licensee failed to perform an adequate design review to ensure the intake structure crane trolley and bridge rail were constructed to seismic class II over I standards. The licensee failed to ensure the intake structure crane trolley rail, trolley rail clip, trolley clip connection, crane rail, crane rail clip and crane clip connection were evaluated for loads due to the safe shutdown earthquake loading concurrent with a lifted load. In response to this issue, the licensee performed an operability determination and concluded that the crane was operable but non-conforming based on a load test that was performed at 1.25 times the rated capacity. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02353.

  The team determined that the failure to perform an adequate design review to ensure the intake structure crane trolley and bridge rail were constructed to seismic class II over I standards was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, the licensee failed to comply with seismic class II over I requirements to ensure the intake structure crane structural integrity when subjected to safe shutdown earthquake loads concurrent with a lifted load; for safe load handling of heavy loads near the safety-related raw water system. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened to Exhibit 4, "External Events Screening Questions," because it was a function specifically design to mitigate a seismic event. Per Exhibit 4 the issue screened to a more detailed risk evaluation because if the seismic function were assumed to be completely failed and a load were dropped it would impact the safety function of the raw water system. Therefore, the Region IV senior reactor analyst performed a more detailed risk evaluation. Given that the frequency of the initiating event is less than 1×10^{-6}, the analyst determined that the finding was of very low safety significance (Green). This finding had a crosscutting aspect in the

area of human performance associated with documentation because the licensee failed to maintain complete, accurate, and up-to-date documentation. [H.7] (Section 1R21.2.12.1)

- Severity Level IV/Green. The team identified two examples of a Severity Level IV, Green, non-cited violation, of 10 CFR 50.59, "Changes, Tests and Experiments," for the licensee's failure to obtain a license amendment prior to implementing a change if the change would result in a departure from a method of evaluation described in the updated safety analysis report. Specifically, on February 23, 2015, and March 10, 2015, the licensee changed the facility to incorporate increased seismic damping for use in the intake structure crane and intake superstructure seismic analysis and seismic design; and in the raw water piping seismic analysis, respectively. In response to this issue, the licensee declared the intake structure as operable but non-conforming pending resolution of a license amendment request to permit the use of the increased damping value; and declared the raw water system as operable but non-conforming pending completion of the corrective actions to determine what actions are necessary to restore compliance to the licensing basis. This finding was entered into the licensee's corrective action program as Condition Reports CR 2015-02224 and CR 2015-02842.

  The team determined that the failure to identify that the proposed change to incorporate increased seismic damping for use in the intake structure crane and intake superstructure seismic analysis and seismic design; and in the raw water piping seismic analysis, was a performance deficiency. This finding was also evaluated using traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences; and there was a reasonable likelihood that the change would have required NRC review and approval prior to implementation. Specifically, the licensee failed to determine that the proposed updated safety analysis report change, and associated design calculations, did involve a change to a structure, systems, or components such that it did adversely affect an updated safety analysis report described design function; less conservative seismic damping values, which required an evaluation to be performed. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. Since the violation is associated with a Green reactor oversight process violation, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d(2) of the NRC Enforcement Policy. This finding had a crosscutting aspect in the area of human performance associated with design margins because individuals failed to ensure margins were carefully guarded and changed only through a systematic and rigorous process. [H.6] (Section 1R21.2.12.2)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, that "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of

design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to March 13, 2015, the licensee did not properly verify the adequacy of the raw water system flow rate to its safety related components through calculational methods or through a suitable testing program. The licensee failed to include the raw water pumps discharge check valves allowable back leakage acceptance criteria into the design calculation. In response to this issue, the licensee performed an operability determination and verified that with the current back leakage flow rates all downstream safety related loads would be properly cooled. This finding was entered into the licensee's corrective action program as Condition Reports CR 2015-01801, and CR 2015-01835.

The team determined that the failure to verify the adequacy of the raw water system design through calculational methods or through a suitable test program was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to have adequate measures in place to ensure that a suitable test program verified design inputs which ensured the design attributes of the raw water system. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance. (Section 1R21.2.13)

- Green. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures." Specifically, prior to February 25, 2015, the licensee failed to follow Procedure FCSG-56, "Time Critical Operation Standard," to ensure all time critical operator actions were validated and verified. In response to this issue, the licensee determined that the continual training of job performance measures that test competency in completing many of the time critical actions provides a basis that all times are achievable. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02443.

  The team determined that the inadequate implementation of Procedure FCSG-56 for validation and verification of operator time critical actions was a performance deficiency. This finding was more than minor because it was associated with the human performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not adequately implement Procedure FCSG-56 to ensure that all operator time critical actions listed in Attachment 1 were properly validated and verified; therefore the licensee could not demonstrate that all operator time critical actions could be executed in accordance with the design basis. In accordance with Inspection Manual Chapter 0609, Appendix A, "The

Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of human performance associated with consistent process because individuals failed to demonstrate an understanding of the decision making process and use it consistently. [H.13] (Section 1R21.4)

Licensee-Identified Violations

A violation of very low safety significance that was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and associated corrective action tracking number is listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

This inspection of component design bases verifies that plant components are maintained within their design basis. Additionally, this inspection provides monitoring of the capability of the selected components and operator actions to perform their design basis functions. As plants age, modifications may alter or disable important design features making the design bases difficult to determine or obsolete. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity cornerstones for which there are no indicators to measure performance.

1R21 Component Design Basis Inspection (71111.21)

.1 Overall Scope

To assess the ability of the Fort Calhoun Station, equipment and operators to perform their required safety functions, the team inspected risk significant components and the licensee's responses to industry operating experience. The team selected risk significant components for review using information contained in the Fort Calhoun Station, probabilistic risk assessments and the U. S. Nuclear Regulatory Commission's (NRC) standardized plant analysis risk model. In general, the selection process focused on components that had a risk achievement worth factor greater than 1.3 or a risk reduction worth factor greater than 1.005. The items selected included components in both safety-related and nonsafety-related systems including pumps, circuit breakers, heat exchangers, transformers, and valves. The team selected the risk significant operating experience to be inspected based on its collective past experience.

To verify that the selected components would function as required, the team reviewed design basis assumptions, calculations, and procedures. In some instances, the team performed calculations to independently verify the licensee's conclusions. The team also verified that the condition of the components was consistent with the design basis and that the tested capabilities met the required criteria.

The team reviewed maintenance work records, corrective action documents, and industry operating experience records to verify that licensee personnel considered degraded conditions and their impact on the components. For selected components, the team observed operators during simulator scenarios, as well as during simulated actions in the plant.

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design basis have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed

performance test results; significant corrective actions; repeated maintenance; Title 10 CFR 50.65(a)1 status; operable, but degraded, conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 15 to 25 total samples that include risk-significant and low design margin components, components that affect the large-early-release-frequency (LERF), and operating experience issues. The sample selection for this inspection was 16 components, 2 components that affect LERF, and 6 operating experience items. The selected components and associated operating experience items supported risk significant functions including the following:

a. Electrical power to mitigation systems: The team selected several components in the electrical power distribution systems to verify operability to supply alternating current (ac) and direct current (dc) power to risk significant and safety-related loads in support of safety system operation in response to initiating events such as loss of offsite power, station blackout, and a loss-of-coolant accident with offsite power available. As such the team selected:

    - 125 Vdc Battery EE-8A and EE-8B (counted as two components)
    - 480 Vac Safety Related Motor Control Center MCC 3B1
    - Emergency Diesel Generator DG-1 and DG-2 Voltage Regulators
    - Raw Water Pump AC-10C Breaker 1A3-10
    - Reactor Protective System

b. Components that affect LERF: The team reviewed components required to perform functions that mitigate or prevent an unmonitored release of radiation. The team selected the following components:

    - Maintenance Hatch O-Rings
    - Shutdown Cooling Containment Isolation Valves HCV 347 and 348

c. Mitigating systems needed to attain safe shutdown: The team reviewed components required to perform the safe shutdown of the plant. As such the team selected:

    - Auxiliary Building Crane
    - Auxiliary Feedwater Pumps FW-6 and FW-10 runout concerns
    - Component Cooling Water Heat Exchanger C Valves HCV 491A and 491B
    - Emergency Diesel Generator Fuel Oil Transfer Pumps FO-37 and FO-57
    - Intake Structure
    - Raw Water Pump AC-10C
    - Raw Water Cooler D Valves HCV 2879A, 2879B, 2883A, and 2883B
    - Shutdown Cooling Heat Exchanger AC-4B

.2 Results of Detailed Reviews for Components

.2.1 125 Vdc Batteries EE-8A and EE-8B

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with 125 Vdc Batteries EE-8A and EE-8B. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Calculations for electrical distribution, system load flow/voltage drop to verify that bus capacity and voltages remained within minimum acceptable limits.
- Sizing calculations to verify input assumptions, design loading, and environmental parameters are appropriate and that the battery cell is sized to perform the battery design basis function.
- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.
- Fort Calhoun's response to Event Notification 49967, Part 21, "Misaligned Separators in LCR-25 Standby Batteries."
- Fort Calhoun's response to NRC Information Notice 2013-05, "Battery Expected Life and Its Potential Impact on Surveillance Requirements."

b. Findings

.1 Failure to Perform an Adequate Battery Sizing and Load Profile Calculation

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to verify or check the adequacy of design of the 125 Vdc batteries. Specifically, the licensee failed to perform an adequate load profile and voltage drop calculation to ensure the batteries were sized to meet their design basis requirements.

Description. The licensee is committed to IEEE 485-1983, "IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations." As a result, in 2009 when the licensee replaced the 125 Vdc batteries they utilized a vendor program which incorporates the requirements of IEEE 485-1983 to validate the design. The EDSA software package performed battery sizing and load flow analysis on the 125 Vdc systems and calculated the required battery size in terms of the number of positive plates needed. The battery profiles for a 4-hour station blackout, an 8-hour design basis accident, and an 8-hour safe shutdown are the license basis for

the batteries as documented in Calculation FC05690, “Battery Load Profile and Voltage Drop Calculation,” Revision 9. These profiles are created based on the loads that will be energized by the batteries during the associated event. The team identified three examples of the licensee’s failure to properly account for various loads to be energized during these profile events. The examples are:

- The licensee failed to include inrush currents for circuit breaker charging spring motors. The licensee assumed the associated load would be 2.5 amps per breaker, and should have included the inrush current of 15 amps per breaker. The licensee re-performed the battery sizing analysis using the correct inrush currents and documented the issue in Condition Report CR 2015-01791.

- The licensee failed to follow IEEE 485-1983 for a load classified as a “random load.” The licensee had identified that main generator disconnect switch DS-T1 could be energized at any point during the event; however the load was added at a convenient period in the profiles. This method was contrary to IEEE 485-1983, which requires random loads to be added to the most critical time in the duty cycle, to simulate the worst case load on the battery. The licensee re-performed the battery sizing analysis treating the disconnect switch as a random load and documented the issue in Condition Report CR 2015-01753.

- The licensee failed to assume the worst case load currents on the 125 Vdc systems when developing the load profiles. Each year the licensee measures the normal running load currents on the 125 Vdc systems, and uses these current values for the load profiles for the batteries. These measured currents do not account for the worst case conditions, such as lower than normal room temperatures, or component degradation due to improper maintenance or age. These worst case load currents will require more current to be drawn from the battery, reducing available battery margin. The licensee performed preliminary calculations assuming a 50 degree Fahrenheit drop in room temperatures for associated components, and re-performed the battery sizing analysis to ensure the batteries can meet the expected design basis loading with these higher current requirements. The licensee documented the issue in Condition Report CR 2015-01825.

These three examples result in an increase in the required current demand and reduce the available margin on the batteries. In December 2014, the licensee evaluated reducing Battery EE-8A aging factor, associated with battery sizing, down from 1.25 to 1.2 to gain additional margin; documented in Condition Report CR 2014-14857. Reducing the aging factor to 1.2 allowed sufficient margin to account for the increased current associated with the three issues identified above. With the reduced aging factor, Battery EE-8A is considered operable but non-conforming until modifications are performed that will regain the available margin and restore the aging factor to 1.25. All of these issues were combined for tracking purposes into Condition Report CR 2014-14857.

Analysis. The team determined that the failure to adequately perform a battery sizing and load profile calculation, to ensure proper battery size and margin was maintained, was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and

adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to account for inrush currents, random loads, and worst case load currents during load profile and battery sizing calculations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis…are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, prior to March 13, 2015, the licensee failed to ensure that design control measures were established that assured the design basis was correctly translated into specifications. Specifically, battery sizing and load profile calculations failed to include proper design data for inrush currents, a random load, and possible worst case load currents. In response to these issues, the licensee updated the design values to account for the missed loads to ensure the batteries maintained adequate available margin. This finding was entered into the licensee's corrective action program as Condition Report CR 2014-14857. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015007-01, "Failure to Perform an Adequate Battery Sizing and Load Profile Calculation."

.2 Failure to Establish Correct Acceptance Criteria Values for Battery Intercell Resistance Measurements

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to ensure the 125 Vdc battery design was correctly translated into procedures. Specifically, the licensee failed to update the acceptance criteria for battery intercell connection resistances, following battery replacement.

Description. The licensee is committed to IEEE 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." As a result, in 2009 when the licensee replaced the 125 Vdc batteries new baseline intercell connection resistances were measured following installation. IEEE 450-1980, which states, in part, "when any intercell connection or terminal connection detail resistance value exceeds its installation value by more than 20 percent, disassemble, clean, reassemble, and retest it."

Battery maintenance Procedures EM-RR-EE-0900, “Inspection and Maintenance for Station Battery No. 1 (EE-8A),” Revision 13a, and EM-RR-EE-0901, “Inspection and Maintenance for Station Battery No. 2 (EE-8B),” Revision 14, provide steps to compare the measured intercell resistance measurements to the baseline resistance values. Step 5.9.5 of EM-RR-EE-0900, and Step 7.9.4 of EM-RR-EE-0901 compare the measured intercell resistance to the “Maximum Resistance” column and states, in part, “If intercell or intertier connection resistances DO NOT meet Acceptance Criteria in the ‘Maximum Resistance’ column, then indicate disassembly and cleaning are required by recording a ‘yes’ in the ‘Cleaning Required’ column of the data sheet.”

During the last performance of the procedure per Work Order WO 00469748-01 (EE-8A), intercell connection 8-9 measured 19.5 micro-ohms. The maximum resistance column value was listed as 24.2 micro-ohms. At the time of battery installation, the installed resistance value for connection 8-9 was recorded as 16 micro-ohms, and therefore the 20 percent acceptance criteria should have been 19.2 micro-ohms. Because the maximum resistance value was for the previously installed battery and not specific to the current battery, the measured value of 19.5 micro-ohms did not exceed the acceptance criteria documented in the procedure. However, it would have been flagged as needing disassembly and cleaning if the correct value of 19.2 micro-ohms was listed. The licensee’s corrective actions included reviewing and comparing all intercell connection resistances to the proper acceptance criteria for both batteries, and performing a visual inspection of the intercell connectors to ensure no adverse condition existed. The licensee documented this issue in Condition Report CR 2015-02129.

Analysis. The team determined that the failure to establish the correct acceptance criteria values for battery intercell resistance measurements was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee had incorrect acceptance criteria for maximum intercell connection resistance measurements, and failed to identify an intercell connection that should have been disassembled, cleaned, reassembled, and remeasured. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 2, “Mitigating Systems Screening Questions,” the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of human performance associated with documentation because the licensee failed to maintain complete, accurate and up-to-date documentation. [H.7]

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “Measures shall be established to assure that applicable regulatory requirements and the design basis…are correctly translated into specifications, drawings, procedures, and instructions.” Contrary to the above, since 2009 the licensee failed to ensure that design control measures were established that assured the design basis was correctly translated into

procedures. Specifically, the licensee failed to update battery maintenance procedures with the current maximum intercell resistance values. In response to this issue, the licensee performed a visual inspection of the battery intercell connections, performed a review of the latest intercell resistance measurements to identify any values that exceeded the correct acceptance criteria value, and performed an immediate operability determination.

This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02129. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015007-02, "Failure to Establish Correct Acceptance Criteria Values for Battery Intercell Resistance Measurements."

.3 Failure to Account for Elevated Battery Room Temperature Effects on Battery Service Life

Introduction. The team identified a Green finding for the licensee's failure to verify or check the adequacy of design of the 125 Vdc batteries from environmental effects. Specifically, the licensee failed to account for the effects of elevated battery room temperature on expected battery service life, in accordance with EPRI Standard TR-100248, "Stationary Battery Guide: Design Application, and Maintenance," Revision 2.

Description. Battery cell design assumes that for maximum battery life the batteries are housed in a nominal environment which equates to a room temperature of 77 degrees Fahrenheit. The battery rooms at Fort Calhoun are consistently measured at 81 and 82 degrees Fahrenheit for Battery EE-8A and EE-8B respectively. Elevated temperatures above the nominal temperature cause the battery to degrade at an accelerated rate, and must be accounted for in the expected service life of the battery.

In Section 3.2.1 of EPRI TR-100248, two methods of calculating expected service life are provided: 1) using the average temperature over a whole year, and 2) using the temperatures taken during monthly surveillance testing for the past year. The licensee's preliminary calculations determined that the expected service life of Battery EE-8A was 17.8 and 17.7 years, respectively for each method; and Battery EE-8B was 16 and 16.1 years, respectively for each method. The licensee sized the batteries for replacement at 20 years, but failed to recognize that the elevated battery room temperatures reduced the expected service life to 17 and 16 years, respectively. The licensee's corrective actions included performing the above service life calculations, evaluating industry actions for ambient temperature degradation, and determining when to increase testing frequency as a result of reduced service life. The licensee documented this issue in Condition Report CR 2015-02390.

Analysis. The team determined that the failure to account for elevated battery room temperature effects on battery service life was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, if left uncorrected, it could lead to a

more significant safety concern in that the batteries could fail to maintain sufficient capacity and go undetected when testing at the normal 5 year interval. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of problem identification and resolution associated with operating experience because the licensee failed to evaluate and implement the EPRI standard based on industry experience when measuring room temperature readings above the optimal battery room temperature. [P.5]

Enforcement. This finding does not involve enforcement actions because no violation of a regulatory requirement was identified. In response to this issue, the licensee performed an immediate operability determination to evaluate the effects of the elevated battery room temperatures and to determine when to modify the testing frequency based on the shorter life of the batteries. This finding was entered into the licensee's corrective action program as Condition Report CR-2015-02390. Because this finding does not involve a violation and was of very low safety significance, it is identified as FIN 05000285/2015007-03, "Failure to Account for Elevated Battery Room Temperature Effects on Battery Service Life."

.2.2 480 Vac Safety Related Motor Control Center MCC 3B1

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with 480 Vac Safety Related Motor Control Center MCC 3B1. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for electrical distribution, system load flow/voltage drop, short-circuit, cable de-rate for fire-wrap, and electrical protection to verify that bus capacity and voltages remained within minimum acceptable limits.

- The protective device settings and circuit breaker ratings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance; including the cable aging management program.

- Results of completed preventative maintenance on switchgear and breakers, including breaker tracking.

b. Findings

No findings were identified.

.2.3 Emergency Diesel Generator DG-1 and DG-2 Voltage Regulators

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Emergency Diesel Generator DG-1 and DG-2 Voltage Regulators. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Engineering Change Modification that replaced Voltage Regulators and resultant design revisions, including work orders and condition reports.
- Vendor manual for replacement Voltage Regulators.
- Procedures for preventive maintenance, inspection, and testing.
- Generator output traces, including observing a field flash of emergency diesel generator DG-2.

b. Findings

No findings were identified.

.2.4 Raw Water Pump AC-10C Breaker 1A3-10

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Raw Water Pump AC-10C Breaker 1A3-10. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for electrical distribution, system load flow/voltage drop, short-circuit, and electrical protection to verify that bus capacity and voltages remained within minimum acceptable limits.

- The protective device settings and circuit breaker ratings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance; including the cable aging management program.

b. Findings

No findings were identified.

.2.5 Reactor Protective System

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the Reactor Protective System. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation, specifically ripple on the dc power supply outputs.

- Vendor and component manufacturer guidance for maintenance and replacement of system power supplies.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

- Fort Calhoun's response to Information Notice 2012-11 "Age Related Capacitor Degradation."

b. Findings

Inadequate Justification for Power Supplies Installed Beyond Vendor Recommended Life

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to adequately evaluate reactor protective system power supplies that are installed beyond the vendor recommended service life. Specifically, the licensee failed to verify or check the adequacy of: 1) service life as a function of expected life minus shelf life; 2) vendor

requirements for in-storage and post-storage maintenance; and 3) including or addressing laboratory failure analysis conclusions that a required component was, although functional, at its "end of life" after 18 years.

Description. The team reviewed Fort Calhoun's response to NRC Confirmatory Action Letter dated September 2, 2011, Item 3.4.2, dealing with replacing reactor protective system power supplies. The licensee's response, "Fort Calhoun Power Supply Evaluation," dated October 31, 2011, contained several items the team could not validate. These items focused on the limiting component for expected life, preventative maintenance activities, and the results of third party laboratory testing on expected life.

The licensee considered the electrolytic capacitor as the bounding component in the power supply's expected life. The vendor recommendation that the service life is equal to the expected life minus shelf life was cited several times in the evaluation, however, an assumption was made that the capacitor expected life equaled the service life of the power supply. The team was concerned that this ignored shelf life in the licensee's warehouse, and potentially at the capacitor manufacturer and/or power supply vendor; which could result in power supplies installed in the plant that are beyond the vendor recommendation for service life.

In addition, the licensee failed to consider vendor recommendations for in-storage preventative maintenance and cautions about placing the power supply in service if in-storage preventative maintenance activities were not performed. The licensee's evaluation listed the vendor recommendation for "reforming" the capacitor, by periodically powering it up, but failed to disposition the lack of performing this activity. Additionally, the vendor cautioned that if the capacitor reforming was not performed, "line voltage should be increased gradually" over a period of time, referred to as a soft start, to prevent damage. This is consistent with the capacitor manufacturer (Sprague) recommendation, "we recommend after extended storage…that capacitors be measured and those that exceed specified initial d-c leakage limits be reconditioned." The licensee assumed the service life was unaffected by the lack of in-storage preventative maintenance and no steps were taken to ensure a soft start to prevent capacitor damage before installation.

Finally, NRC Confirmatory Action Letter Item 3.4.2 required the licensee to include remnant life based on stress testing of removed power supplies. The licensee sent removed power supplies to a third-party laboratory for analysis; "Omaha Public Power District Power Supply Investigation," by Exponent, March 2012. The laboratory reported some power supplies functional, but at "end of life" due to capacitors, with little to no remnant life at 18 years. The licensee, however, assumed the reported functionality as indicative of continued life, and did not include or address reported lack of remnant life of the capacitor, therefore failing to justify service life beyond vendor recommendations. The licensee's immediate corrective actions included performing an engineering evaluation to support an operable but non-conforming condition and generating rework activities to replace/refurbish the installed power supplies.

Analysis. The team determined that the failure to perform an adequate justification for having reactor protective system power supplies installed beyond vendor recommend life was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and

adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to perform an adequate justification for continued operation for reactor protective system power supplies that were beyond vendor recommended life. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, prior to March 13, 2015, the licensee failed to verify or check the adequacy of design of the replacement reactor protective system power supplies that were installed beyond vendor recommend life. Specifically, the licensee failed to verify or check the adequacy of: 1) service life as a function of expected life minus shelf life; 2) vendor requirements for in-storage and post-storage maintenance; and 3) including or addressing laboratory failure analysis conclusions that a required component was, although functional, at its "end of life" after 18 years. In response to this issue, the licensee performed an immediate operability determination and verified the power supply's ripple checks were still within tolerance. This finding was entered into the licensee's corrective action program as Condition Reports CR 2015-02809 and CR 2015-02811. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: 05000285/2015007-04, "Inadequate Justification for Power Supplies Installed Beyond Vendor Recommended Life."

.2.6 Maintenance Hatch O-Rings

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Maintenance Hatch O-Rings. The team also conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Engineering Change Package “Equipment Hatch Cover (AE-1) Blocking Device” to verify that the design change would not result in a degradation of the capability of the equipment hatch to seal the containment.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

b. Findings

No findings were identified.

.2.7 Shutdown Cooling Containment Isolation Valves HCV 347 and 348

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Shutdown Cooling Containment Isolation Valves HCV 347 and 348. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Calculations for motor-operated valve design basis operating conditions, degraded voltage, torque and thrust capabilities, and weak-link analyses.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

b. Findings

No findings were identified.

.2.8 Auxiliary Building Crane

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, selected drawings and calculations, maintenance and test procedures, and condition reports associated with the Auxiliary Building Crane. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

- Design basis calculations for crane uprate from 75 tons to 106 tons. Including the licensee's compliance with ASME NOG-1-2004, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," 2004.

b. Findings

Failure to Perform an Adequate Evaluation for the Auxiliary Building Crane

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to perform an adequate design review to upgrade the auxiliary building single failure proof crane capacity. Specifically, the licensee failed to comply with ASME NOG-1-2004, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)."

Description. Fort Calhoun Station Updated Safety Analysis Report Section 14.24 specifies the auxiliary building crane meets the requirements of ASME NOG-1-2004 as a single failure proof system. In 2009 the licensee received a license amendment to upgrade the auxiliary building crane from a 75 ton to a 106 ton maximum rated capacity. During a review of the upgrade design Calculation, FC07263, "Auxiliary Building Crane (HE-2) Uprate from 75 Ton to 106 Ton Maximum Rated Capacity," Revision 0, the team identified three examples where the licensee failed to meet the requirements of ASME NOG-1-2004 as a single failure proof system. The examples are:

- The licensee failed to have an adequate analysis for the trolley rail, crane rail and trolley/crane rail splice connections for the applied stresses due to load combinations specified in ASME NOG-1-2004 Section 4140. The licensee documented these issues in Condition Reports CR 2015-02232 and CR 2015-02673.

- The licensee incorrectly designed the crane runway rail clips to inelastic acceptance limits. ASME NOG-1-2004 Section 4153 stipulates the crane seismic analysis to be linear elastic. In addition, the applied tensile loading on the rail clip was incorrectly calculated assuming two rail clips resist the load instead of one rail clip. The licensee documented this issue in Condition Report CR 2015-02319.

- The licensee incorrectly assumed that sliding would occur at the crane rail/wheel interface thus limiting the applied loads to frictional forces. This assumption resulted in significantly reduced seismic loads and was inconsistent with the boundary condition requirements stipulated in ASME NOG-1-2004 Section 4153. The licensee documented this issue in Condition Report CR 2015-02718.

As a result, prior to March 13, 2015, the licensee's design basis calculation was not sufficient to ensure conformance with the requirements for safe load handling of heavy loads over safety-related structures, systems, or components; such as the spent fuel pool and the safety injection refueling water storage tank. The licensee performed some immediate calculational assessments to allow the crane to be used for loads less than

the maximum 106 tons; limited the use of the main hook to the original 75 ton value until the long term actions, including calculational revisions and field modifications can be completed to restore the crane to fully operable; and declared the crane operable but non-conforming. All of these issues were combined for tracking purposes into Condition Report CR 2015-02718.

Analysis. The team determined that the failure to perform an adequate design review to upgrade the auxiliary building single failure proof crane capacity was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, the licensee failed to comply with ASME NOG-1-2004 requirements to ensure the auxiliary building crane remained elastic when subjected to design loads for safe load handling of heavy loads. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened to Exhibit 4, "External Events Screening Questions," because it was a function specifically design to mitigate a seismic event. Per Exhibit 4 the issue screened to a more detailed risk evaluation because if the seismic function were assumed to be completely failed and a load were dropped it would impact the spent fuel pool cooling or the safety injection refueling water storage tank functions. Therefore, the Region IV senior reactor analyst performed a more detailed risk evaluation.

The analyst noted that the auxiliary building crane was originally qualified to conduct lifts of up to 75 tons. Therefore, only lifts with weight greater than 75 tons were of concern to this evaluation. The licensee provided documentation of 6 lifts that had been performed with greater than 75 tons over the 6-year period from 2009 to 2015. This represented an average of one lift with 1 hour per lift (P_{Lift}) over the Exposure period of 1 year.

Using the methods described in the Risk Assessment of Operational Events Handbook, Volume 2, "External Events," Revision 1.01, the analyst determined that the frequency of a seismically-induced loss of offsite power at Fort Calhoun Station ($\lambda_{seismic\text{-}LOOP}$) was 1.01×10^{-4}/year. Given the six successful lifts and the characteristics of low energy earthquakes, the analyst assumed that earthquakes that were not large enough to cause a loss of offsite power would not negatively impact the functioning of the auxiliary building crane. Therefore, the analyst calculated the frequency of a seismic event occurring with strength large enough to result in a loss of offsite power ($\lambda_{seismic\text{-}drop}$) as follows:

$$\lambda_{seismic\text{-}drop} = \lambda_{seismic\text{-}LOOP} * P_{Lift} * \text{Exposure}$$

$$= 1.01 \times 10^{-4}/\text{year} * 1/\text{year} * 1/8760 \text{ years}$$

$$= 1.16 \times 10^{-8}/\text{year}$$

Given that the frequency of the initiating event is less than 1×10^{-6}, the analyst determined that the finding was of very low safety significance (Green).

The team determined that this finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, prior to March 13, 2015, the licensee failed to perform an adequate design review to upgrade the auxiliary building single failure proof crane capacity. Specifically, the licensee failed to comply with ASME NOG-1-2004, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)." In response to this issue, the licensee performed an operability determination and concluded that the crane was operable but non-conforming, and limited the use of the main hook to the original 75 ton value until the long term actions can be completed to restore the crane to fully operable. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02718. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015007-05, "Failure to Perform an Adequate Evaluation for the Auxiliary Building Crane."

.2.9 Auxiliary Feedwater Pumps FW-6 and FW-10 runout concerns

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Auxiliary Feedwater Pumps FW-6 and FW-10 runout concerns. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Corrective Action documents issued in the past 5 years to verify that repeat failures, and potential chronic issues, will not prevent the steam-driven and motor-driven pumps and associated components from performing their safety function.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance. The team put special emphasis on testing methodology, the values assigned to acceptance criteria, and whether the values supported design parameters and assumptions.

- Runout analysis including design assumptions, limiting parameters, and whether the available net positive suction head was sufficient to satisfy the required net

positive suction head to prevent cavitation and assure the capability of each pump to perform its safety function.

- Fort Calhoun's response to Event Notification Report 45724 "Auxiliary Feedwater Pumps May Experience Runout at Low Steam Generator Pressures."

b. Findings

Failure to Perform an Adequate Auxiliary Feedwater Pump Runout Design Calculation

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to verify or check the adequacy of design of the Auxiliary Feedwater system. Specifically, the licensee failed to have an adequate design calculation and a suitable testing program for the turbine-driven pump runout analysis.

Description. The team identified that the licensee failed to adequately verify the turbine-driven auxiliary feedwater pump would be able to perform its design function under all design basis conditions. Specifically, the licensee failed to analyze the effects of several parameters that negatively impact the runout analysis of the turbine-driven pump (FW 10). Calculation FC08310, "Auxiliary Feedwater Motor Driven Pump FW-6 and Turbine Driven Pump FW-10 Performance and Runout Evaluation," assumed that the lowest possible steam generator pressure was the limiting value for this analysis. No basis for this assumption was documented in the calculation and it neglected to consider that a greater steam generator pressure would provide a greater motive force to the turbine and therefore the pump. Based on this assumption Calculation FC08310 determined that the net positive suction head required, associated with a steam generator pressure of 460 psig, was 52.5 feet and the available net positive suction head was 52.7 feet.

The team questioned the validity of using the lowest steam generator pressure of 460 psig, as the limiting case for pump runout conditions and the required net positive suction head. As a result of these concerns, the licensee contacted the original equipment manufacturer of the turbine-driven pump and determined that a steam generator pressure of 700 psig was significantly more limiting. After reevaluating these parameters, the licensee determined that at this steam generator pressure the required net positive suction head would be about 70 feet.

The team also identified that Calculation FC08310 did not include any penalty for allowable back flow through the discharge check valve of the idle motor-driven pump (FW-6) and isolation Valve HCV-1384. The surveillance procedure's acceptance criteria of up to a 20 psi increase in pressure at Pump FW-6 discharge line allows for potentially significant reverse flow; the pressure increase is the result of reverse flow through the closed discharge check valve. Valve HCV-1384 separates the safety related and non-safety related portions of the auxiliary feedwater system. These pathways decrease the available net positive suction head for pump FW-10 and were not accounted for in the calculation.

These errors resulted in the licensee incorrectly determining that the turbine-driven pump had sufficient net positive suction head to satisfy all design basis scenarios. The

licensee's corrective actions included performing an operability determination; revising several calculational errors, including removing conservatisms which resulted in a gain of net positive suction head; and determining that less than 90 percent of the required net positive suction head was available, which violates the design basis of the pump. The licensee contacted the original equipment manufacturer and was provided with a testing summary that determined the turbine-driven pump could operate for up to 600 seconds with only 70 percent of the required net positive suction head. This provided the licensee with the basis for an operable but non-conforming condition until long term corrective actions, including evaluating a modification to the auxiliary feedwater system, could be completed. The licensee captured these concerns in the corrective action program as Condition Report CR 2015-02414. The team determined that engineering had recently reviewed the runout calculation and did not identify these errors as a result of not having a conservative bias.

Analysis. The team determined that the failure to verify the adequacy of the auxiliary feedwater system design through calculational analysis and a suitable test program was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to have adequate measures in place to ensure an acceptable design analysis and a suitable test program to verify the design inputs and ensure the capability of the auxiliary feedwater system to perform its safety function. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of human performance associated with conservative bias because individuals failed to use decision making practices that emphasize prudent choices over those that are simply allowed. [H.14]

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, prior to March 13, 2015, the licensee did not verify the adequacy of design of the auxiliary feedwater system. Specifically, the licensee failed to have an adequate design calculation and a suitable testing program for the turbine-driven pump runout analysis. In response to this issue, the licensee contacted the original equipment vendor and determined that the pump would still operate outside its design basis. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02414. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a

non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015007-06, "Failure to Perform an Adequate Auxiliary Feedwater Pump Runout Design Calculation."

.2.10 Component Cooling Water Heat Exchanger C Valves HCV 491A and 491B

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Component Cooling Water Heat Exchanger C Valves HCV 491A and 491B. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Corrective Action documents issued in the past 5 years to verify that repeat failures, and potential chronic issues, will not prevent the component cooling water heat exchanger valves from performing their safety function.
- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

b. Findings

No findings were identified.

.2.11 Emergency Diesel Generator Fuel Oil Transfer Pumps FO-37 and FO-57

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Emergency Diesel Generator Fuel Oil Transfer Pumps FO-37 and FO-57. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Calculations for fuel oil tank capacity, fuel oil inventory requirements, and net positive suction head requirements.
- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

b. Findings

No findings were identified.

.2.12 Intake Structure

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, selected drawings and calculations, procedures, and condition reports associated with the Intake Structure. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Historical correspondence detailing the design and construction of the building.
- Intake structure crane seismic class II over I design calculations.
- Corrective actions associated with previous NRC violations, including reconstituted design calculations that demonstrate compliance with 10 CFR Part 50, Appendix A, "General Design Criteria."
- Fort Calhoun's disposition of Licensee Event Report 2013-012, "Intake Structure Crane Seismic Qualification."
- Fort Calhoun's disposition of Unresolved Item 2013012-10, "Unverified Design for Seismic Damping Values for Raw Water Piping Inside the Turbine Building."

b. Findings

.1 Failure to Perform an Adequate Evaluation for the Intake Structure Crane Trolley and Bridge Rail

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to perform an adequate design review to ensure the intake structure crane trolley and bridge rail were seismic class II over I constructed. Specifically, the licensee failed to ensure the intake structure crane trolley rail, trolley rail clip, trolley clip connection, crane rail, crane rail clip and crane clip connection were evaluated for loads due to the safe shutdown earthquake loading concurrent with a lifted load.

Description. During a review to close Licensee Event Report 2013-012, "Intake Structure Crane Seismic Qualification," the team identified the following statement of concern:

> However, damage to the unprotected fire protection headers that exist in the intake structure are not considered in the load drop analysis. Therefore, this

piping may be damaged during a seismic event if HE-5 is in use during a seismic event. The volume of this flooding that could be produced by this event is outside of the assumptions of the intake structure internal flooding analysis and could result in all four raw water pumps becoming inoperable.

The licensee documented this concern in Condition Report CR 2013-15474 and initiated actions to evaluate the intake structure crane in accordance with the seismic class II over I requirements for the safe shutdown earthquake load concurrent with a lifted load. The analysis included the overhead crane placed at several locations throughout the intake structure during a safe shutdown earthquake event while carrying a lifted load corresponding to an equivalent weight of the circulating water pump motor (maximum expected load for the intake structure crane).

The team reviewed the licensee's actions to address the discrepancy. In particular Condition Report CR 2013-15474 and Calculation FC07811, "Intake Structure Overhead Crane (HE-5) Seismic II/I Qualification," Revision 1, and identified that the licensee did not have an analysis for the trolley rail, trolley rail clip, trolley clip connection, crane rail, crane rail clip and crane clip connection for the applied stresses due to the safe shutdown earthquake concurrent with the lifted load of the circulating water pump motor. The intake structure crane trolley rail, trolley rail clip, and trolley clip connection are the load transfer elements from the trolley to the intake structure crane bridge girder; while the crane rail, crane rail clip, and crane clip connection are the load transfer elements from the crane bridge girder to the intake crane support structure.

As a result, prior to March 13, 2015, the design basis calculation was not sufficient to ensure conformance with seismic class II over I requirements for safe load handling of heavy loads over safety-related structures, systems, or components; such as the raw water system. The licensee's immediate corrective actions included declaring the intake structure crane operable but non-conforming, verifying no work requiring the crane use was planned for the near future, and initiating actions to update the design calculations and implement modifications to the intake structure crane to restore to a fully operable condition. The licensee captured this issue in the corrective action program as Condition Report CR 2015-02353. See Section 4OA3 for closure of the license event report.

Analysis. The team determined that the failure to perform an adequate design review to ensure the intake structure crane trolley and bridge rail were constructed to seismic class II over I standards was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to events to prevent undesirable consequences. Specifically, the licensee failed to comply with seismic class II over I requirements to ensure the intake structure crane structural integrity when subjected to safe shutdown earthquake loads concurrent with a lifted load; for safe load handling of heavy loads near the safety-related raw water system. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened to Exhibit 4, "External Events Screening Questions," because it was a function specifically design to mitigate a seismic event. Per Exhibit 4 the issue screened to a more detailed risk evaluation because if the seismic function were assumed to be completely failed and a load were dropped it would

degrade one or more trains of the raw water system, a system that supports risk significant systems and functions. Therefore, the Region IV senior reactor analyst performed a more detailed risk evaluation.

The analyst noted that the intake structure crane trolley and bridge rail has been operated throughout the life of the plant and is likely capable of maintaining its integrity during low energy earthquakes with or without minimal loads on the hook. Therefore, only lifts of pumps, motors, strainers and associated equipment were of concern to this evaluation. The licensee provided documentation of lifts associated with repairs of the circulating water pumps and motors, the raw water pumps, motors and strainers, and the screen wash strainers. The analyst calculated a total of 220 hours that large loads were on the hook of the crane over the 16-year period from 1999 to 2015. This represented an average of 13.75 hours of lifts (P_{Lift}) over the Exposure period of 1 year.

Using the methods described in the Risk Assessment of Operational Events Handbook, Volume 2, "External Events," Revision 1.01, the analyst determined that the frequency of a seismically-induced loss of offsite power at Fort Calhoun Station ($\lambda_{seismic\text{-}LOOP}$) was 1.01×10^{-4}/year. Given the 220 hours of successful lifts and the characteristics of low energy earthquakes, the analyst assumed that earthquakes that were not large enough to cause a loss of offsite power would not negatively impact the functioning of the intake structure crane. Therefore, the analyst calculated the frequency of a seismic event occurring with strength large enough to result in a loss of offsite power ($\lambda_{seismic\text{-}drop}$) as follows:

$$\lambda_{seismic\text{-}drop} = \lambda_{seismic\text{-}LOOP} * P_{Lift} * \text{Exposure}$$

$$= 1.01 \times 10^{-4}/\text{year} * 13.75 \text{ hours/year} * 1/\text{year}/8760 \text{ hours/years}$$

$$= 1.59 \times 10^{-7}/\text{year}$$

Given that the frequency of the initiating event is less than 1×10^{-6}, the analyst determined that the finding was of very low safety significance (Green).

This finding had a crosscutting aspect in the area of human performance associated with documentation because the licensee failed to maintain complete, accurate and up-to-date documentation. [H.7]

<u>Enforcement</u>. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, prior to March 13, 2015, the licensee failed to perform an adequate design review to ensure the intake structure crane trolley and bridge rail were constructed to seismic class II over I standards. Specifically, the licensee failed to ensure the intake structure crane trolley rail, trolley rail clip, trolley clip connection, crane rail, crane rail clip and crane clip connection were evaluated for loads due to the safe shutdown earthquake loading concurrent with a lifted load. In response to this issue, the licensee performed an operability determination and concluded that the crane was operable but non-conforming based on a load test that was performed at 1.25 times the rated capacity.

This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02353. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015007-07, "Failure to Perform an Adequate Evaluation for the Intake Structure Crane Trolley and Bridge Rail."

.2 Failure to Obtain Prior NRC Approval for a Change in Seismic Analysis Damping

Introduction. The team identified two examples of a Severity Level IV, Green, non-cited violation, of 10 CFR 50.59, "Changes, Tests and Experiments," for the licensee's failure to obtain a license amendment prior to implementing a change if the change would result in a departure from a method of evaluation described in the updated safety analysis report. Specifically, the licensee changed the facility to incorporate increased seismic damping for use in the intake structure crane and intake superstructure seismic analysis and seismic design; and in the raw water piping seismic analysis.

Description. The licensee is permitted to make changes to the facility as described in the updated safety analysis report without prior NRC approval, provided that these changes do not result in a departure from a method of evaluation described in the updated safety analysis report and used in establishing the plant design bases. Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," states that the methods described in Nuclear Energy Institute NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59. Nuclear Energy Institute NEI 96-07, Section 4.3.8, states that licensees can make changes to elements of a methodology without first obtaining a license amendment if the results are essentially the same as, or more conservative than, previous results. The team concluded that the change, an increase in seismic damping values resulted in a less conservative result for the two examples discussed below.

The first example was identified on February 23, 2015, during the team's review of Calculations FC07810, "Intake Superstructure Seismic II/I Qualification," Revision 1, and FC07811, "Intake Structure Overhead Crane (HE-5) Seismic II/I Qualification," Revision 1. The team identified that the licensee used incorrect damping values in the seismic analysis of the intake structure crane and intake superstructure. The licensee used 7 percent for the maximum hypothetical earthquake in lieu of the Updated Safety Analysis Report, Appendix F, Table F-2 damping value of 1 percent for welded steel assemblies and/or 2 percent for bolted steel assemblies. While not specifically approved for Fort Calhoun, the NRC has approved the use of 7 percent damping values in Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," for bolted steel structures. In response to this issue, the licensee declared the intake structure as operable but non-conforming pending resolution of a license amendment request to permit the use of the increased damping value. The licensee documented this issue in the corrective action program as Condition Report CR 2015-02224.

The second example was identified on March 10, 2015, during the team's review to close Unresolved Item 05000285/2013012-10, "Unverified Design for Seismic Damping Values for Raw Water Piping Inside the Turbine Building." The team reviewed Calculation FC02400, "Input Data Corresponding to Stress Analysis RW-111A and

Qualification Summary," Revision 4. The raw water piping subsystem RW-111A runs from the component cooling water heat exchangers in the auxiliary building to the discharge tunnel connection in the turbine building. On Page 2.8 of Calculation FC02400 it stated, in part:

> "Therefore pending NRC approval of Turbine Building spectra, PVRC spectra developed in References 5.4.3, Figures A-33 and A-34, for SSE Turbine Building conditions were compared to design basis OBE spectra."

The damping and spectra curve developed by the Pressure Vessel Research Council (PVRC) was ASME Code Case N-411, "Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1." The Updated Safety Analysis Report, Appendix F, damping for piping is not described as Pressure Vessel Research Council damping and was 0.5 percent for the design basis earthquake and maximum hypothetical earthquake condition. In addition, NRC Safety Evaluation Report of Alternate Seismic Criteria and Methodologies - Fort Calhoun Station, dated April 16, 1993, was incorporated by reference into Updated Safety Analysis Report, Appendix F.

The Safety Evaluation Report of Alternate Seismic Criteria and Methodologies required Regulatory Guide 1.61 damping and prohibited the use of ASME Code Case N-411 for the seismic analysis of piping. Fort Calhoun inappropriately used ASME Code Case N-411, for which they were specifically not approved. In response to this issue, the licensee declared the raw water system as operable but non-conforming pending completion of the corrective actions to determine what actions are necessary to restore compliance to the licensing basis. This NRC identified violation closes (URI 05000285/2013012-10, "Unverified Design for Seismic Damping Values for Raw Water Piping Inside the Turbine Building"). See Section 4OA5.3 for closure of the URI. The licensee documented this issue in the corrective action program as Condition Report CR 2015-02842.

Analysis. The team determined that the failure to identify that the proposed change to incorporate increased seismic damping for use in the intake structure crane and intake superstructure seismic analysis and seismic design; and in the raw water piping seismic analysis, was a performance deficiency. This finding was also evaluated using traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences; and there was a reasonable likelihood that the change would have required NRC review and approval prior to implementation. Specifically, the licensee failed to determine that the proposed updated safety analysis report change, and associated design calculations, did involve a change to a structure, systems, or components such that it did adversely affect an updated safety analysis report described design function; less conservative seismic damping values, which required an evaluation to be performed. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that

did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. Since the violation is associated with a Green reactor oversight process violation, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d(2) of the NRC Enforcement Policy. This finding had a crosscutting aspect in the area of human performance associated with design margins because individuals failed to ensure margins were carefully guarded and changed only through a systematic and rigorous process. [H.6]

Enforcement. The team identified two examples of a Severity Level IV, Green, non-cited violation, of 10 CFR 50.59, "Changes, Tests and Experiments," Section (c)(2) which states, in part, that a licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would have resulted in a departure from a method of evaluation described in the updated safety analysis report used in establishing the design bases or in the safety analyses. Contrary to the above, on February 23, 2015, and March 10, 2015, the licensee failed to obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, test, or experiment that would result in a departure from a method of evaluation described in the updated safety analysis report. Specifically, the licensee changed the facility as described in the updated safety analysis report to incorporate increased seismic damping for use in the intake structure crane and intake superstructure seismic analysis and seismic design; and in the raw water piping seismic analysis. In response to this issue, the licensee declared the intake structure as operable but non-conforming pending resolution of a license amendment request to permit the use of the increased damping value; and declared the raw water system as operable but non-conforming pending completion of the corrective actions to determine what actions are necessary to restore compliance to the licensing basis. Because this violation was entered into the corrective action program as Condition Reports CR 2015-02224 and CR 2015-02842, 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 (NCV), consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000285/2015007-08, "Failure to Obtain Prior NRC Approval for a Change in Seismic Analysis Damping."

.3 (Opened) Unresolved Item URI 05000285/2015007-09, "Intake Structure Design Requirements"

Introduction. The team reviewed correspondence and design calculations related to the intake structure and identified several issues of concern, predominately related to the design of the structure for flooding and a barge impact.

Description. The following issues were discussed during the inspection, but the licensee was unable to provide the required information to be able to disposition the issue in accordance with Inspection Manual Chapter 0612 as either minor or more than minor. Additionally, the aforementioned calculations were performed by third-party vendors, so the licensee staff did not have specific knowledge of the calculations because they did

not prepare them. To close the unresolved item the NRC needs additional information to be able to address the following concerns:

- Hydrodynamic forces and levee construction: The team found that the intake structure licensing basis design requirements are unclear for the natural phenomenon expected during a flood, specifically hydrodynamic effects. The team questioned the rationale that the design of the intake structure only included the consideration of hydrostatic forces during a design basis flood of the Missouri River. The expected design would include hydrodynamic forces (as the river is moving at a velocity of approximately 10 feet per second) and the potential impact loads from debris floating downstream. Historical correspondence between the Atomic Energy Commission (predecessor to the NRC) and the licensee related to this question revealed that the original flood mitigation strategy included construction of temporary earthen levees (EA10-032, Revision 0; Preliminary Safety Analysis Report, Supplement 12; and Final Safety Analysis Report, Supplement 15, February 1972). Designing the intake structure to only include hydrostatic loads from the Missouri River during flood stages is not intuitive; however, it can be reasoned that the licensee's original proposed solution of constructing earth levees to protect the plant would be sufficient to isolate the flowing river water and debris such that hydrodynamic flooding effects would not need to be considered. The licensee does not construct temporary earthen levees to protect structures during flood events, nor does it have plans to do so in the future.

- Code limits for flooding conditions: Calculation FC07802, "Intake Structure Project Design Manual," states that the NRC "accepted" Calculation FC01414, "Exterior Wall Design, Diesel Generator Room," Revision 1, for the exterior wall designs of the auxiliary building and that the design code limits established in that calculation would be used in the reconstitution of the intake structure. Calculation FC01414, Revision 1, was not retrievable during the inspection. Revision 0, dated March 23, 1968, was found and provided to the team, which showed the building was designed using ultimate strength design limits of the American Concrete Institute ACI 318 code for a flood to elevation 1014 MSL. However, the final safety analysis report requires the intake structure and auxiliary building to be designed to the working stress design limits of the American Concrete Institute ACI 318 code. In addition, Revision 0 identifies an area that is overstressed during a flood event and would require larger diameter reinforcement bars than analyzed. The licensee was unable to provide the team with documentation explaining how the ultimate stress design versus working stress design and the additional reinforcement bars were resolved.

- Required loading combinations: A design memo (WIP file 019070, dated July 7, 1970) between the architect engineer who designed the intake structure and the licensee states that "the intake structure has been designed as Class I structure up to El. 1007'-6" but that the perimeter wall between elevations 1007'-6" and 1014'-6" is not so designed and will not resist simultaneous flood and earthquake loading…" Final Safety Analysis Report, Section 5.11.3, requires the intake structure to be able to resist a simultaneous flood and earthquake loading combination. The licensee was unable to provide the team with documentation explaining why the intake structure perimeter walls

were not designed to resist the required loading combination of an earthquake and flood.

- Building, foundation, and soil stiffness: Calculations FC07495, “Intake Structure Seismic Analysis and Pile Design,” and FC07803, “Intake Structure, Sub-Structure Analysis,” derived new stiffness values for both the steel pile foundation that supports the intake structure and the soil stiffness interaction values. The use of these values affects the behavior of the structure during an earthquake and barge impact. In addition, there are discrepancies in the natural frequency of the building that are not addressed. The licensee was unable to provide the team with documentation explaining the use of the new stiffness values or the lack of addressing the natural frequency of the building.

- Sloshing: Calculation FC07495 assumes that the opposing walls of the intake structure are “relative short distances” so the water acts as a rigid body (no water sloshing effects are accounted for). This assumption did not have any technical justification. Specifically, a representative raw water cell is 20 feet deep by 10 feet long by 7 feet wide, and the total estimated water weight for each cell is over 1.1 million pounds. Consequently sloshing effects could comprise a large additional load on the structure. The licensee was unable to provide the team with documentation explaining the lack of including water sloshing effects on the walls of the intake structure.

- Barge impact loading: A barge crashing into the intake structure is a required design load for the building. Previous NRC Inspection Reports (05000285/2005011 and 05000285/2009006) documented the lack of design calculations for this required loading condition. Calculation FC07803 analyzes the barge impact loads, however the team raised the following concerns which the licensee was unable to provide documentation explaining the discrepancies:

    - A barge impact is only assumed to occur at average river levels (986.5 to 1001.3 feet MSL) and is neglected at low or high river levels (979.5 to 1014 feet MSL), because of the assumption that the Army Corps of Engineers would impose barge restrictions or raise/lower water level. No formal agreement or justification for this assumption has been provided.

    - The barge impact calculation arbitrarily selected a 40 percent increase in material strengths of the intake structure. This increased factor was based on “engineering experience” without any technical justification or data to support the use of higher strength materials than specified in the final safety analysis report.

    - The barge impact assumes a barge velocity of 2.2 feet per second; however the actual river velocity is 10 feet per second. The calculation assumes the maximum barge speed is 10 feet per second, despite the fact the barge has propulsion and could travel at a speed equal to the river velocity plus the barge propulsion speed. The use of a lower velocity increases the duration of the loading event; however it significantly underestimates the energy of the barge impact. If the barge

is assumed to impact the intake structure at 10 feet per second, the calculation states that:

> …the force would entirely overwhelm the flexural capacity of the nosing wall sections, which confirms that these structural elements would fail under Barge Impact loading conditions. However, the extremely short duration of the postulated loading condition (0.075 seconds) may not be structurally significant on the remainder of the Intake Structure, nor on the safety-related SSCs.

The team questioned why the lower velocity was used, because the effective force of a 10 feet per second impact (distributed over a short duration) would be significantly greater than a 2.2 feet per second impact distributed over a larger time duration.

.2.13 Raw Water Pump AC-10C

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Raw Water Pump AC-10C. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Corrective Action documents issued in the past 5 years to verify that repeat failures, and potential chronic issues, will not prevent the raw water pump and associated components from performing their safety function.
- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance. The team put special emphasis on testing methodology, the values assigned to acceptance criteria, and whether the values supported design parameters and assumptions.

b. Findings

.1 Failure to Adequately Account for Raw Water Pump Discharge Check Valve Back Leakage

Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to verify or check the adequacy of design of the raw water system. Specifically, the licensee failed to include allowable back leakage acceptance criteria for the raw water pumps discharge check valves into the design calculation.

Description. The team identified that the licensee failed to adequately verify the raw water system would be able to perform its design function under all design basis conditions. The team discovered that Calculation FC07259, "FCS RW/CCW Gothic Model – Additional Cases," Revision 3, did not account for back leakage through the check valves. The failure to include the back leakage in the raw water system calculation resulted in overestimation of the coolant flow to safety related components.

In addition, the licensee did not have a suitable test program to determine the amount of back leakage associated with the surveillance procedure acceptance criteria. The surveillance procedure's acceptance criteria of up to a 10 percent pressure change ensured no gross leakage past the check valve but did not correlate the pressure change with a leakage flow rate. As a result of the team's finding, it was determined that the reverse flow rate associated with 10 percent could be as large as 1500 gpm. This large loss of flow rate was not accounted for in system modeling and hydraulic calculation to ensure the downstream safety related components would be properly cooled. In fact, the calculation did not account for any loss of flow due to back leakage through the check valve.

The licensee's corrective actions included performing an immediate operability determination and reviewing completed surveillance test acceptance criteria on the raw water pumps discharge check valves. The actual reverse flow rates were minimal; ensuring that all downstream safety related loads would be properly cooled. The licensee captured these concerns in the corrective action program as Condition Reports CR 2015-01801 and CR 2015-01835. The team determined that this finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Analysis. The team determined that the failure to verify the adequacy of the raw water system design through calculational methods or through a suitable test program was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to have adequate measures in place to ensure that a suitable test program verified design inputs which ensured the design attributes of the raw water system. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. The team determined that this finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, that "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational

methods, or by the performance of a suitable testing program." Contrary to the above, prior to March 13, 2015, the licensee did not properly verify the adequacy of the raw water system flow rate to its safety related components through calculational methods or through a suitable testing program. Specifically, the licensee failed to account for the potential back leakage through the raw water pumps discharge check valves. In response to this issue, the licensee verified that with the current back leakage flow rates to all downstream safety related loads would be properly cooled. This finding was entered into the licensee's corrective action program as Condition Reports CR 2015-01801 and CR 2015-01835. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015007-10, "Failure to Adequately Account for Raw Water Pump Discharge Check Valve Back Leakage."

.2 (Opened) Unresolved Item URI 05000285/2015007-11, "Raw Water Strainer Analysis and Commercial Dedication"

Introduction. The team reviewed Calculation FC07716, "Commercial Grade Dedication of Raw Water Strainers 12A and 12B," which qualifies the non-safety related raw water strainer system as ASME safety related equipment via the commercial dedication process. The team found that although the calculation was assigned a Fort Calhoun calculation number, it had not been through the owner acceptance review and contained items that required further inspection.

Description. The following issues were discussed during the inspection, but the licensee was unable to provide the required information to be able to disposition the issue in accordance with Inspection Manual Chapter 0612 as either minor or more than minor. Additionally, parts of the aforementioned calculation were performed by third-party vendors, so the licensee staff did not have specific knowledge of the calculation because they did not prepare them. To close the unresolved item the NRC needs additional information to be able to address the following concerns:

- Material Properties: Calculation FC07716 performed a chemical analysis to conclude the strainer was a cast iron material because there was no specific receipt paperwork on the design or construction. The chemical analysis concluded that the strainer body was cast iron but the test equipment used was unable to measure carbon. The team questioned the validity of using equipment unable to detect carbon as carbon is a major component of steel and its content has a significant impact on its properties. Especially with steel being a major component in a power plant. In addition, the calculation used a hardness test to correlate the strength of the cast iron. The calculation used average hardness values, rather than minimum values as is required by ASME codes, to correlate a strength value. The team questioned why the average value was used instead of the minimum specified.

- Analysis methodology and results: The calculation used ANSYS simulation software to compute the stress levels within the equipment. However, the results of the computer model were not verified or validated. Specifically, the team was not provided with evidence that parametric or sensitivity studies were performed – particularly for the model "mesh." The ANSYS model had a coarse

mesh size, with less than eight elements through the thickness. Typically a mesh this coarse will not accurately predict the stress levels. Additionally, the ANSYS results showed some areas with stress levels that exceeded the acceptance criteria established with the calculation (another issue of concern), but the high stress was averaged across the thickness. The ANSYS model has several assumptions that the licensee could not provide a basis for; therefore the team questioned the validity of the models results for commercial dedication.

- Acceptance criteria: The purpose of Calculation FC07716 is to convert the non-safety related commercial equipment into safety related by commercial dedication. However, the analysis uses the ASME design code as the acceptable stress limits because the raw water strainers are part of the raw water piping system. The final safety analysis report states that the raw water system piping code of record is United States of America Standards USAS B31.1. The team questioned why ASME allowable stresses were used instead of United States of America Standard USAS B31.1 allowable stress limits. No code reconciliation was provided during the inspection to address the issue.

- Loading conditions: Calculation FC07716 does not address a barge impact loading condition. As stated in URI 05000285/2015007-09 (see Section 1R21.2.12.3), a barge impact is a required loading condition for the intake structure. The barge impact is a large loading and will produce large accelerations within the intake structure. In turn, these accelerations will affect the strainer system, which is assumed to be made from cast iron – which is a brittle material. The effects of the barge impact on the raw water strainer system do not appear to be analyzed.

.2.14 Raw Water Cooler D Valves HCV 2879A, 2879B, 2883A, and 2883B

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Raw Water Cooler D Valves HCV 2879A, 2879B, 2883A, and 2883B. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

- Corrective Action documents issued in the past 5 years to verify that repeat failures, and potential chronic issues, will not prevent raw water cooler valves and associated components from performing their safety function.

- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance; including the aging management program as a result on being in the period of extended operation.

b. Findings

No findings were identified.

.2.15 Shutdown Cooling Heat Exchanger AC-4B

a. Inspection Scope

The team reviewed the updated safety analysis report, system description, design basis documents, the current system health report, selected drawings and calculations, maintenance and test procedures, and condition reports associated with Shutdown Cooling Heat Exchanger AC-4B. The team also performed walkdowns and conducted interviews with system engineering personnel to ensure the capability of this component to perform its desired design basis function. Specifically, the team reviewed:

- Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.
- Calculations for the structural evaluation of the heat exchanger due to a shutdown cooling mode pressure increase, cyclic fatigue, and tube plugging allowances.
- Procedures for preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

b. Findings

No findings were identified.

.3 Results of Reviews for Operating Experience

.3.1 Inspection of Event Notification Report 45724 "Auxiliary Feedwater Pumps May Experience Runout at Low Steam Generator Pressures"

a. Inspection Scope

The team reviewed the licensee's evaluation of Event Notification Report 45724 "Auxiliary Feedwater Pumps May Experience Runout at Low Steam Generator Pressures" to verify the licensee's corrective actions to address the concerns described in the event notification. This event notification discusses a condition at Fort Calhoun where both auxiliary feedwater pumps were declared inoperable due to an evaluation that determined that the pumps may experience runout at low steam generator pressures. Further review by the licensee determined that the pumps remained operable and the event notification was retracted. The team verified that the licensee's review adequately addressed the issues discussed in the event notification.

b. Findings

See Section 1R21.2.9 for the enforcement aspects of the review of this event notification.

.3.2 Inspection of Event Notification Report 49967 “Part 21 Misaligned Separators in LCR-25 Standby Batteries”

a. Inspection Scope

The team reviewed the licensee’s evaluation of Event Notification Report 49967 “Part 21 Misaligned Separators in LCR-25 Standby Batteries” to verify the licensee performed an applicability review and took corrective actions, if appropriate, to address the concerns described in the event notification. This event notification discusses misaligned separators between positive and negative plates on batteries provided by C&D Technologies. The team verified that the licensee’s review adequately addressed the issues in the event notification.

b. Findings

No findings were identified.

.3.3 Inspection of NRC Generic Letter 88-17 “Loss of Decay Heat Removal”

a. Inspection Scope

The team reviewed the licensee’s evaluation of Generic Letter 88-17 “Loss of Decay Heat Removal” to verify the licensee performed an applicability review and took corrective actions, if appropriate, to address the concerns described in the generic letter. This generic letter discusses deficiencies that may exist in procedures, hardware, and training in the areas of prevention of accident initiation, mitigation of accidents before they potentially progress to core damage, and control of radioactive material if a core damage accident should occur related to the loss of decay heat removal capability. The team verified that the licensee’s review adequately addressed the issues in the generic letter. However, the team did note that the licensee has not implemented a modification (EC 42649) it initiated in 2008, which was to remove the automatic closure interlock from shutdown cooling isolation valves HCV-347 and HCV-348, related to this generic letter. The licensee currently has an action, documented in Condition Report CR 2014-10365, to determine if the modification is still needed.

b. Findings

No findings were identified.

.3.4 Inspection of NRC Information Notice 2012-11 “Age Related Capacitor Degradation”

a. Inspection Scope

The team reviewed the licensee’s evaluation of Information Notice 2012-11 “Age Related Capacitor Degradation” to verify the licensee performed an applicability review and took corrective actions, if appropriate, to address the concerns described in the information notice. This information notice discusses age related degradation of capacitors that results from epoxy insulation hardening and cracking over time, this allows for a high flow of current and excessive heating. The team verified that the licensee’s review adequately addressed the issues in the information notice.

b. Findings

No findings were identified.

.3.5 Inspection of NRC Information Notice 2013-05 “Battery Expected Life and Its Potential Impact on Surveillance Requirements”

a. Inspection Scope

The team reviewed the licensee’s evaluation of Information Notice 2013-05 “Battery Expected Life and Its Potential Impact on Surveillance Requirements” to verify the licensee performed an applicability review and took corrective actions, if appropriate, to address the concerns described in the information notice. This information notice discusses actions to be taken if battery design margin is exceeded due to load additions or reducing the battery aging factor in order to gain margin on the batteries. The team verified that the licensee’s review adequately addressed the issues in the information notice.

b. Findings

No findings were identified.

.3.6 Inspection of NRC Regulatory Issue Summary 2010-06 “Inservice Inspection and Testing Requirements of Dynamic Restraints (Snubbers)”

a. Inspection Scope

The team reviewed the licensee’s evaluation of Regulatory Issue Summary 2010-06 “Inservice Inspection Testing Requirements of Dynamic Restraints (Snubbers)” to verify the licensee performed an applicability review and took corrective actions, if appropriate, to address the concerns described in the regulatory issue summary. This regulatory issue summary discusses the requirements for the inservice inspection and testing of dynamic restraints (snubbers) under 10 CFR 50.55a(g) and 10 CFR 50.55a(b)(3)(v). The team verified that the licensee’s review adequately addressed the issues in the regulatory issue summary.

b. Findings

No findings were identified.

.4 Results of Reviews for Operator Actions

a. Inspection Scope

The team selected risk-significant components and operator actions for review using information contained in the licensee’s probabilistic risk assessment. This included components and operator actions that had a risk achievement worth factor greater than two or Birnbaum value greater than 1E-6.

For the review of operator actions, the team observed operators during simulator scenarios associated with the selected components as well as observing simulated actions in the plant.

The selected operator actions were:

- Scenario 1, Part 1: The scenario was designed to cause an extended station blackout condition. The crew would first enter the EOP-00, "Standard Post Trip Actions" and then transition to EOP-07, "Station Blackout." The crew would be required to perform operator time critical actions per EOP/AOP Attachment MVA-24, "Minimizing DC Loads."

- Scenario 1, Part 2: The scenario used time compression to move the crew two hours later in the station blackout event timeline. At this point Attachment MVA-24 would be completed in its entirety.

- In-plant job performance measure #1: This job performance measure was designed for a plant operator to perform the field actions for minimizing dc loads in accordance with Attachment MVA-24.

- In-plant job performance measure #2: This job performance measure was designed for a plant operator to align the fire water system as a backup to the raw water system for cooling the component cooling water heat exchangers in accordance with AOP-18, "Loss of Raw Water," Attachment B.

- In-plant job performance measure #3: This job performance measure was designed for a plant operator to transfer diesel fuel oil from storage tank FO-10 to storage tank FO-1 in accordance with EOP/AOP Attachment MVA-19, "Emergency Diesel Generator Long Term Actions."

b. Findings

Failure to Properly Implement Procedures for Verifying Operator Time Critical Actions

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 perform activities affecting quality as prescribed by documented procedures of a type appropriate to the circumstances and accomplished in accordance with these procedures. Specifically, the team identified the failure of the licensee to ensure all time critical operator actions could be performed within the required time per Procedure FCSG-56, "Time Critical Operation Standard."

Description. From February 10-25, 2015, the team observed two crews perform the station blackout simulator scenario. The crews performed the required operator time critical actions within the allotted time. The team requested documentation of the operator time critical actions, in accordance with Procedure FCSG-56, to ensure the station blackout operator time critical actions had been validated. The licensee could not provide documentation that the operator time critical actions had been previously performed within the required times for a station blackout. Specifically, EOP/AOP Attachment MVA-24, "Minimizing DC Loads," seventy five minute requirement for dc

load shedding steps. Based on the two operating crews performance the licensee believed that this action could be completed as expected until the time action could be validated for all operating crews.

Procedure FCSG-56 requires operator time critical actions listed in Attachment 1 be validated to ensure they can be performed within their required action times. The actions listed in Attachment 1 are credited in the licensee's design basis to ensure that the station can satisfy all of its design basis accidents. Without verification and validation of time critical operator actions the licensee cannot be certain that damage to structures, systems, or components would not occur before placing the plant in a stable shutdown condition.

After additional review, the team determined that the licensee could not provide the required documentation to support validation and verification for several other operator time critical actions. The licensee initiated Condition Report CR 2015-02443 to document the operable but non-conforming condition. The licensee's immediate corrective actions included determining that all of the time critical tasks are trained on during requalification training and the majority are informally tracked as part of requalification scenarios, emergency response drills, and classroom training. In addition, the licensee determined that the continual training of job performance measures that test competency in completing many of the time critical actions provides a basis that all times are achievable.

Analysis. The team determined that the inadequate implementation of Procedure FCSG-56 for validation and verification of operator time critical actions was a performance deficiency. This finding was more than minor because it was associated with the human performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not adequately implement Procedure FCSG-56 to ensure that all operator time critical actions listed in Attachment 1 were properly validated and verified; therefore the licensee could not demonstrate that all operator time critical actions could be executed in accordance with the design basis. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding had a crosscutting aspect in the area of human performance associated with consistent process because individuals failed to demonstrate an understanding of the decision making process and use it consistently. [H.13] Procedure FCSG-56 is a quality procedure that prescribes how the licensee is to perform and document time critical operator actions.

Enforcement. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented procedures of a type

appropriate to the circumstances and shall be accomplished in accordance with these procedures." Contrary to the above, prior to February, 25, 2015, the licensee failed to ensure that activities affecting quality as prescribed by documented procedures of a type appropriate to the circumstances were accomplish in accordance with those procedures. Specifically, the licensee failed to follow Procedure FCSG-56 to ensure all time critical operator actions were validated and verified. In response to this issue, the licensee determined that the continual training of job performance measures that test competency in completing many of the time critical actions provides a basis that all times are achievable. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-02443. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: 05000285/2015007-12, "Failure to Properly Implement Procedures for Verifying Operator Time Critical Actions."

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

4OA2 Problem Identification and Resolution (71152)

Component Design Basis Review

The team reviewed action requests associated with the selected components, operator actions, and operating experience notifications. Any related findings are documented in prior sections of this report. However, the team noted the licensee's engineering staff continues to experience challenges with the implementation of the operability determination processes. These challenges include failures to recognize degraded or nonconforming conditions, failures to promptly engage other departments when outside expertise is needed to determine whether a degraded or nonconforming condition meets the criteria for operability, and failures to recognize the appropriate authority for making operability determinations. The team noted that these challenges affected the station's ability to identify problems at a low threshold and to promptly correct conditions adverse to quality.

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

These activities constitute completion of one event follow-up sample, as defined in Inspection Procedure 71153.

(Closed) Licensee Event Report 05000285/2013-012-00, "Intake Structure Crane Seismic Qualification"

On August 2, 2013, while in Mode 5, the licensee identified that the intake structure crane seismic analysis assumes that the crane is not in operation with the hood retracted and in the parked position. The seismic analysis did not evaluate the crane's ability to withstand a seismic event when carrying a lifted load. The licensee's investigation identified that the crane was used when the raw water pumps were required to be operable. The heavy loads drop analysis had previously evaluated the potential damage of a dropped load event that could be caused by a seismic or tornado

initiating event, and showed that the crane would not cause damage to the intake structure. However, damage to the unprotected fire protection headers that exist in the intake structure were not considered in the heavy loads drop analysis. The fire protection headers could be damaged if the crane is carrying a lifted load during a seismic event. The volume of flooding that could be produced is outside of the assumptions of the intake structure internal flooding analysis and could result in all four raw water pumps becoming inoperable. The team reviewed the root cause investigation, procedures, corrective action documents and interviewed station personnel. The enforcement aspects of this violation are discussed in Section 1R21.2.12.1. This licensee event report is closed.

4OA5 Other Activities

.1 Followup of Confirmatory Action Letters or Orders

a. Inspection Scope

On December 2, 2013, the licensee committed by letter (ADAMS Accession No. ML13336A785) to perform a series of actions "for sustained improvement" following restart from an extended outage. These commitments were confirmed by the NRC in a CAL issued December 17, 2013 (ML13351A395).

The team selected a sample of licensee activities that were indicative of the actions the licensee committed to accomplish as confirmed in the December 17, 2013, CAL. The team reviewed whether the licensee took corrective actions as described and whether these corrective actions were effective in addressing the issues that necessitated the issuance of the confirmatory action letter. The actions inspected by the team were those listed in Enclosure 3 of the licensee's December 2, 2013, letter associated with Key Driver 4 (Design and Licensing Basis Control and Use), and the Cables and Connections of Key Driver 7 (Equipment Performance).

b. Observations and Assessments

1. Design and Licensing Basis

The team reviewed the licensee actions to which it committed under Key Driver 4 in Enclosure 3 to its December 2, 2013, letter. The commitment associated with Key Driver 4 (identified in NRC Inspection Report 2014009 as PIIM item 4.a) AI 2013-17439-003, "Ensure Design Engineering performs at least one engineering self-assessment on a risk significant system in 2014," is closed. The team determined that the licensee adequately performed one engineering self-assessment in 2014.

Additionally, the team reviewed the commitments associated with Key Driver 4 (identified in NRC Inspection Report 2014009 as PIIM item 4.a) AI 2013-05570-026, AI 2013-05570-076, and AI 2013-05570-093 which deals with defining the current licensing basis, assuring the design basis documents remain current and complete, and validating that the design and licensing basis has been translated into plant operation. The team reviewed the licensee's first pilot system for design basis reconstitution, the raw water system.

The team expressed some concerns about the licensee's method for performing the reconstitution and the self-checks built into the program to ensure the licensee was achieving the expected outcome. For example, the program schedule shared with the team contained a critical system review to ensure that the licensee was achieving the desired results, but it was scheduled to be performed before all of the inputs to the system were complete. This would result in a critical system review of a system that has not had all its design aspects completed. However, the team did note that the independent oversight committee raised some concerns that the licensee is in the process of incorporating to ensure a better finished product.

As a result, the licensee is making changes to the reconstitution effort program. The team determined that these commitments should remain open until more items have been reviewed to be able to determine if the licensee is effectively addressing the issues that necessitated the action item.

2. Cables and Connections

The team reviewed the licensee actions to which it committed under Key Driver 7 in Enclosure 3 to its December 2, 2013, letter. The commitments associated with Key Driver 7 (identified in NRC Inspection Report 2014009 as PIIM item 7.g) AI 2012-8134-026, AI 2009-04216-020, AI 2013-17441-001, and AI 2013-17441-002 are closed. The team determined that the licensee had adequately addressed the concerns that necessitated the action items.

c. Findings

No findings were identified.

.2 Containment Internal Structure Safety Injection Tank Beams 22A and 22B Interferences (East Side) Modification

a. Inspection Scope

The team reviewed Engineering Change EC58236, "Containment Internal Structure Columns (East Side)," which was done in order to reconstitute the containment internal structure design at elevation 1013' and bring that portion of containment internal structure into compliance with the Fort Calhoun Station Updated Safety Analysis Report. This modification involved temporary or permanently relocating mechanical (piping) and electrical (cables, conduits) commodities in order to accommodate installation of the structural column to support Beam 22A and 22B. The team reviewed design calculations associated with the relocation of the mechanical and electrical commodities.

b. Findings

No findings were identified.

.3 (Closed) Unresolved Item 05000285/2013012-10, "Unverified Design for Seismic Damping Values for Raw Water Piping Inside the Turbine Building"

During the 2013 Special Inspection, see Inspection Report 05000285/2013012, the team identified an unresolved item involving the damping used for the raw water piping seismic analysis for the portion that is routed inside the turbine building. Specifically, the team was concerned that the licensee had not properly evaluated the use of Pressure Vessel Research Council damping.

During this inspection, the team reviewed additional information provided by the licensee regarding the design and licensing basis of the raw water system, specifically the portion routed through the turbine building. The team reviewed calculations, procedures, corrective action documents and interviewed station personnel. The enforcement aspects of this violation are discussed in Section 1R21.2.12.2. Based on this review, the team resolved the concerns discussed and unresolved item 05000285/2013012-10 is closed.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On March 13, 2015, the inspectors presented the inspection results to Mr. L. Cortopassi, Vice President and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as a non-cited violation.

- Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "Measures shall be established to assure that applicable regulatory requirements and the design basis…are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, in December 2014, the licensee identified a failure to ensure that the design basis for station blackout was correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to include a diesel fuel oil pump load in the design basis battery load profiles for station blackout, design basis accident, and safe shutdown. This finding was determined to be of very low safety significance because the licensee performed an operability determination and concluded that Battery EE-8A was operable but non-conforming; by reducing the aging factor from 1.25 to 1.2. The licensee entered this issue into their corrective action program as Condition Report CR 2014-14857.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Anderson, Manager, Engineering Projects
M. Bakhit, Design Engineer
D. Bonwell, Supervisor, Training
C. Cameron, Regulatory Assurance
S. Campagna, Exelon
L. Cortopassi, Site Vice President
J. Denton, Design Engineer
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M. Frans, Manager, Projects
H. Goodman, Site Engineering Director
C. Hooker, Design Engineer
J. Kelly, Design Engineer
M. Kluge, Senior Staff Engineer
T. Leibel, Design Engineer
S. Linguist, Operations
K. Maassen, Program Engineer
E. Matzke, Regulatory Assurance
D. Motte, Design Engineer
A. Noseir, Senior Design Engineer
S. Salgia, Exelon
M. San, Exelon
T. Simpkin, Manager, Regulatory Assurance
C. Sterba, Design Engineer
M. Swan, System Engineer
S. Swanson, Operations Director
K. Wells, Design Engineer
Z. Wineinger, Design Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000285/2015007-09	URI	Intake Structure Design Requirements (Section 1R21.2.12.3)
05000285/2015007-11	URI	Raw Water Strainer Analysis and Commercial Dedication (Section 1R21.2.13.2)

Opened and Closed

05000285/2015007-01	NCV	Failure to Perform an Adequate Battery Sizing and Load Profile Calculation (Section 1R21.2.1.b.1)
05000285/2015007-02	NCV	Failure to Establish Correct Acceptance Criteria Values for Battery Intercell Resistance Measurements (Section 1R21.2.b.2)

Opened and Closed

05000285/2015007-03	FIN	Failure to Account for Elevated Battery Room Temperature Effects on Battery Service Life (Section 1R21.2.b.3)
05000285/2015007-04	NCV	Inadequate Justification for Power Supplies Installed Beyond Vendor Recommended Life (Section 1R21.2.5)
05000285/2015007-05	NCV	Failure to Perform an Adequate Evaluation for the Auxiliary Building Crane (Section 1R21.2.8)
05000285/2015007-06	NCV	Failure to Perform an Adequate Auxiliary Feedwater Pump Runout Design Calculation (Section 1R21.2.9)
05000285/2015007-07	NCV	Failure to Perform an Adequate Evaluation for the Intake Structure Crane Trolley and Bridge Rail (Section 1R21.2.12.1)
05000285/2015007-08	NCV	Failure to Obtain Prior NRC Approval for a Change in Seismic Analysis Damping (Section 1R21.2.12.2)
05000285/2015007-10	NCV	Failure to Adequately Account for Raw Water Pump Discharge Check Valve Back Leakage (Section 1R21.2.13)
05000285/2015007-12	NCV	Failure to Properly Implement Procedures for Verifying Operator Time Critical Actions (Section 1R21.4)

Closed

05000285/2013-012-00	LER	Intake Structure Crane Seismic Qualification (Section 4OA3)
05000285/2013012-10	URI	Unverified Design for Seismic Damping Values for Raw Water Piping Inside the Turbine Building (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

Calculations

Number	Title	Revision/Date
EA10-059	Safety Classification Analysis of Fuel Oil Transfer Pump FO-37 and Supporting Components	0
EA12-015	Cables and Connections	0
EA90-066	Maximum MCC Loadings and Incoming Feeder Analysis	8
EA91-007	4160V/480V Short Circuit Analysis	6
EA91-084	Breaker/Fuse Coordination Study	8, 8B, 8C, 8D
EA91-142	480V Breaker Setpoint Determination	1
EA92-072	Diesel Generator Loading Transient Analysis Using Paladin Design Base 4.0	8
EA-FC-90-076	Cable Tray Loading Calculation	11
EA-FC-92-080	Resolution of MOV Operating Conditions Design Basis Discrepancies	7

Calculations

Number	Title	Revision/Date
EA-FC-99-007	Fort Calhoun Station 4160V/480V Short Circuit Current Analysis; Using EDSA Design Base 3.0	6
FC01015	Stress Analysis for Sub-system SI-073C	3
FC01414	Exterior Wall Design, Diesel Generator Room	0
FC02051	Design of Pipe Support WDS-160	2
FC02485	Calc-Input & Qualification Data Corresponding To Stress Analysis SI-073C	2
FC03158	Calc – Construction Structural Design of Containment Parametric Studies, Pipe Pile	August 29, 1967
FC04990	MCC-3A1, 3B1, and 3C1 Feeder Cable Derating Due to Fire Wrap	1
FC05337	Determination of Startup Feedwater Pump Diesel Driver Exhaust Line Size and Fuel Oil Transfer Pump Suction Line, Discharge Line, and Overflow Line	0
FC05386	Calculation of CCW System Heatup During a DBA LOCA if Raw Water System is Unavailable	0
FC05659	Development of Flow Coefficients for the Raw Water and Component Cooling Water System Analysis	3
FC05690	Battery Load Profile and Voltage Drop Calculation	8, 9, 9M, 9N, 9O
FC05694	Calculation of Minimum Reactor Coolant Time Using Shutdown Cooling System	3
FC05829	MOV Degraded Voltage Calculation, Using EDSA Design Base 3.0	12
FC05876	An Analysis of OPPD, FCS Motor Operated Actuator's Torque/Thrust Capabilities and Motor Vendor Operated Valve's Thrust/Torque Requirements	17
FC06184	Seismic and Weak Link Analysis Summary for HCV-347. MOV Group 2	0
FC06203	MOV Testing – Instrument Errors, Setpoint Calculation	5
FC06323	IPEEE and USI A-46 Seismic Review Floor Response Spectra	0
FC06726	RW Discharge Header Temperature – Direct Cooling Mode	1
FC06871	Diesel Generator Fuel Inventory	1
FC07012	External Missiles due to Tornado Winds and Turbine Generator Overspeed	0

Calculations

Number	Title	Revision/Date
FC07235	Structural Evaluation of Shutdown Cooling Heat Exchangers for Shutdown Cooling Mode Pressure Increase	0
FC07259	FCS RW/CCW Gothic Model – Additional Cases	3
FC07262	Auxiliary Building Crane (HE-2) Support Structure Evaluation	0
FC07263	Auxiliary Building Crane (HE-2) Uprate from 75 Ton to 106 Ton Maximum Rated Capacity	0
FC07272	The Feasibility of Using Fire Water for Cooling the Component Cooling Water System	8
FC07495	Intake Structure Seismic Analysis and Pile Design	1
FC07536	FW-6 and FW-10 Suction and Discharge Piping Friction Loss	0
FC07696	Fort Calhoun AOV Categorization Report	0
FC07716	Commercial Grade Dedication of Raw Water Strainers 12A and 12B	0
FC07802	Intake Structure Project Design Manual	1
FC07803	Intake Structure Design Basis Reconstitution	2
FC07807	Hand Pump Manual Transfer of Diesel Fuel Oil From FO-10 to FO-1	0
FC07808	Intake Structure, Sub-Structure Analysis	1
FC07809	Tornado Differential Pressure for the Class I Structures of the Intake Structure	0
FC07810	Intake Superstructure Seismic II/I Qualification	1
FC07811	Intake Structure Overhead Crane (HE-5) Seismic II/I Qualification	1
FC08072	Seismic Analysis of Raw Water Pumps	1
FC08260	Tornado Missile Protection for Auxiliary Building Elevation 1057' Roof Above Room 81 Blow-Off Panels	0
FC08278	Seismic Evaluation of the Circulating Water Piping in Raw Water Pump Room	0
FC08310	Auxiliary Feedwater (AFW) Motor Driven Pump FW-6 and turbine Driven Pump FW-10 Performance and Runout Evaluation	0
FC08391	Evaluation of Supports ACS-506 and ACS-507 for CIS East Side Interferences	0
FC08392	Evaluation of Supports for Conduits EC12831A and EC12831B for CIS East Side Interferences	0

Calculations

Number	Title	Revision/Date
FC08394	Evaluation of Support WDH-386 for CIS East Side Interferences	0
FC08405	AXSP-16 Seismic Qualification	0

Condition Reports (CRs)

2005-04345	2010-03921	2012-11039	2013-08034	2014-04310
2008-05686	2011-00650	2012-15755	2013-09502	2014-04971
2008-06133	2011-01774	2012-16746	2013-10976	2014-05785
2009-03769	2011-02603	2012-16857	2013-12724	2014-06140
2009-03977	2011-04821	2012-18025	2013-13741	2014-09499
2009-04216	2011-05053	2012-19444	2013-15474	2014-09782
2009-05180	2011-05995	2013-00273	2013-16870	2014-10062
2009-05617	2011-06111	2013-00514	2013-17191	2014-10123
2009-05912	2011-06815	2013-00607	2013-17439	2014-10365
2009-06145	2011-09526	2013-01705	2013-17441	2014-11001
2009-06438	2011-09713	2013-02087	2013-18306	2014-11580
2009-06615	2012-01452	2013-02933	2013-19018	2014-12117
2009-06616	2012-01825	2013-03695	2013-19170	2014-13249
2010-00220	2012-01946	2013-03866	2013-19502	2014-14519
2010-00883	2012-03544	2013-04609	2013-19907	2014-14762
2010-02446	2012-06115	2013-04863	2013-20715	2014-14857
2010-02561	2012-08134	2013-04864	2013-22090	2014-15103
2010-02754	2012-08617	2013-05570	2014-02591	2014-15105
2010-02793	2012-08684	2013-05678	2014-03079	2015-00766
2010-02800	2012-09117	2013-06060	2014-03338	2015-02486
2010-03762	2012-09928	2013-06455	2014-04232	

Condition Reports (CRs) Generated during the Inspection

2015-00504	2015-01781	2015-01886	2015-02342	2015-02718
2015-00554	2015-01785	2015-01901	2015-02344	2015-02787
2015-01482	2015-01791	2015-01963	2015-02352	2015-02791

2015-01515	2015-01793	2015-02001	2015-02353	2015-02803
2015-01523	2015-01795	2015-02002	2015-02371	2015-02809
2015-01524	2015-01799	2015-02041	2015-02390	2015-02811
2015-01551	2015-01801	2015-02052	2015-02391	2015-02812
2015-01578	2015-01805	2015-02057	2015-02398	2015-02813
2015-01657	2015-01817	2015-02129	2015-02399	2015-02814
2015-01679	2015-01824	2015-02196	2015-02400	2015-02833
2015-01729	2015-01825	2015-02224	2015-02414	2015-02841
2015-01731	2015-01832	2015-02232	2015-02431	2015-02842
2015-01744	2015-01835	2015-02245	2015-02443	2015-02862
2015-01753	2015-01858	2015-02270	2015-02444	2015-02914
2015-01755	2015-01861	2015-02296	2015-02504	2015-02924
2015-01763	2015-01862	2015-02301	2015-02514	2015-02925
2015-01769	2015-01868	2015-02314	2015-02537	2015-03158
2015-01774	2015-01875	2015-02319	2015-02611	
2015-01777	2015-01878	2015-02320	2015-02629	
2015-01778	2015-01883	2015-02330	2015-02673	

Drawings

Number	Title	Revision/Date
0223R0455, Sh. 2	Internal Device Diagrams	0
0223R0455, Sh. 17	Bus No. 1A3 Power & Control Circuit Unit 1A3-10 Raw Water Pump No. AC-10C	3
161F597, Sh. 8	Elementary Diagram AI-30A	27
161F597, Sh. 9	Elementary Diagram AI-30A	24
161F598, Sh. 8	Elementary Diagram AI-30B	20
161F598, Sh. 9	Elementary Diagram AI-30B	21
11405-E-7, Sh. 1	480 Volt Primary Plant Motor Control Center One Line Diagram	68
11405-E-8, Sh. 1	125 Volt DC Misc Power Distribution Diagram P&ID	63
11405-E-73, Sh. 1	Switchgear Diesel Generator & Electrical Penetration	111

Drawings

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11405-M-253, Sh. 1	Steam Generator Feedwater and Blowdown P&ID	99
11405-M-253, Sh. 4	Steam Generator Feedwater and Blowdown P&ID	53
11405-M-254	Composite Flow Diagram Condensate P&ID	56
11405-M-254, Sh. 2	Condensate P&ID	41
11405-M-262, Sh. 1	Fuel Oil Flow Diagram P&ID	63
11405-S-2	Containment Structure Steel Liner	15
11405-S-15	Containment Structure Foundation Mat Reinforcement	6
11405-S-17	Reactor Plant Basement Floor Plan El. 994'-0"	7
11405-S-22	Reactor Plant Sections and Details	9
11405-S-24	Reactor Plant Sections and Details	4
11405-S-38	Reactor Plant Basement Floor Sections and Details	3
11405-S-43	Reactor Plant Reactor Foundation & Fuel Pit Reinforcement	2
11405-S-45	Reactor Plant Steam Generator Compartments Reinforcement	2
11405-S-67	Auxiliary Building Column Schedule & Crane Runway	9
11405-S-313	Intake Structure and Tunnels Plan at Elev. 1007'-6" and Details	19
80055, Sh. 1	12' Dim. x 23' O.A.L. Diesel Fuel Oil Tank FO-1	9
D-1468, Sh 11136	Battery Rack Layout	August 4, 2013
D-4612, Sh. 1	Piping Isometric for FEL Oil Piping between Fuel Tanks FO-10 and FO-38	3
D-4615	480V Motor Control Center MCC – 4C6 One Line Diagram and Elevation View P & ID	10
E-23866-210-130, Sh. 1	Safety Injection and Containment Spray System Flow Diagram P & ID	114
FIG 8.1-1	Simplified One Line Diagram Plant Electrical System P&ID	145
FIG 8.1-1 Sh. 654	Simplified One Line Diagram Plant Electrical System P&ID	141
M-8515, Sh. 1	"L" Two-Step EP Cat III Racks	3
M-10270, Sh. 1	Battery Arrangement, 2 Step EP3 & 2 Tier EP3 (2) Sets of (58) LCR-31 Cells	0

Design Basis Documents

Number	Title	Revision
SDBD-AUX-502	Auxiliary Building	19
SDBD-DG-112	Emergency Diesel Generators	31
SDBD_EE-201	AC Distribution	24
SDBD-SI-130	Shutdown Cooling	23
SDBD-SI-CS-131	Containment Spray	34

Design Change Packages

Number	Title	Revision/Date
EC11123	Station Battery Replacement	March 4, 1993
EC35300	Diesel Generator Voltage Regulator Replacement	0
EC39382	Equipment Hatch Cover (AE-1) Blocking Device	1
EC41956	DG Engine Controls Upgrade	0
EC50088	New Tag Number for Portable Manual Diesel Fuel Oil Transfer Pump and Revise Affected Documents for Calculation FC07807	0
EC58236	Containment Internal Structure Column Interferences (East Side)	0
EC62828	Modify Screen Wash Piping in RW Vault	0
EC62956	DG Room Heater VA-50F	0
EC65261	Design Adequacy of FO-10 to FO-1 Fuel Oil Transfer System	0
ECN-92-10	MOV Torque Switch Limiter Plates	July 19, 1993

Miscellaneous

Number	Title	Revision/Date
	AC-1B Total Fouling Resistance Trend	December 27, 2014
	Exponent OPPD Power Supply Investigation	March 2012
#92-02	Limitorque Technical Update	October 9, 1992
09-023-001-QC-00	Commercial Grade Dedication of Raw Water (RW) Strainers 12A and 12B	0

Miscellaneous

Number	Title	Revision/Date
1763	AC-4B: Disassemble, Clean, Inspect, and Perform ECT	0
18-128	Fort Calhoun Heat Exchanger AC-1B	January 10, 2015
20-69-4332	CCW Heat Exchanger Specification Sheet	March, 11, 1969
45836	Byron Jackson Certified Head Capacity Curve for FW-6	1
602977-MPS-7CALC-003	Limitorque Selection Procedure	5
67037	Flowserve Auxiliary Feedwater Head Capacity Curve for FW-6	0
AC-RW	Raw Water System Component Reclassification Report	0
CA-245A	FW-10 Variable Speed Head Capacity Efficiency RPM curves	0
CE NPSD-548	Requirements for the Removal of the Shutdown Cooling Suction Valve Auto-Closure Interlock	0
EE-0003 Spreadsheet	Monthly/Quarterly Surv. Test Data for Station Battery #1 (EE-8A) – EM-ST-EE-0003; Date Range 10/13/1992 to 12/15/2014	
EE-0004 Spreadsheet	Monthly/Quarterly Surv. Test Data for Station Battery #2 (EE-8B) – EM-ST-EE-0004; Date Range 2/10/1993 to 12/15/2014	
ESA-105	Calculation for conductor Temperatures for Power Cables	August 4, 1986
LIC-13-0061	Exigent License Amendment Request 13-02 Revise Current Licensing Basis to Adopt a Revised Design Basis / Methodology for Addressing Design-Basis Tornado / Tornado Missile Impact	July 21, 2013
LIC-13-0104	Reply to NRC Request for Additional Information (RAI) Regarding Exigent License Amendment Request 13-02 Revise Current Licensing Basis to Adopt a Revised Design Basis / Methodology for Addressing Design-Basis Tornado / Tornado Missile Impact	July 24, 2013
LIC-88-1106	Response to NRC Generic Letter 88-17	January 4, 1988
LIC-89-045	Response to NRC Generic Letter 88-17	February 10, 1989
NED-14-008 DEN	Memorandum from R. L. Church to File FC07259	January 14, 2014

Miscellaneous

Number	Title	Revision/Date
NP-6408	EPRI Guidelines for Establishing, Maintaining, and Extending the Shelf Life Capability of Limited Life Items	1
NP-7410	EPRI Molded Case Circuit Breaker	1
NRC-13-0089	Fort Calhoun Tornado Missile Protection Request for Additional Information (MF2469)	July 23, 2013
NRC-13-0092	RE: Fort Calhoun Tornado Missile Protection Request for Additional Information (MF2469)	July 24, 2013
NRC-13-0095	Fort Calhoun Station, Unit No.1 - Issuance of Exigent Amendment Re: Revise Current Licensing Basis for Addressing Design-Basis Tornado/Tornado Missile Impact (TAC No. MF2469)	July 26, 2013
NUREG 1801	GALL Report	0, 2
OE2010-01199	RIS-10-06 Inservice Inspection And Testing Requirements Of Dynamic Restraints (Snubber)	
OE2012-01452	Response to IN 2012-11, Age Related Capacitor Degradation	
OE2014-00363	C&D Technologies, Inc. (Part 21) – Event #49967	
OE2014-00405	C&D Technologies, Inc. (Part 21) – Log No. 2014-17-00	
OE2014-0818	IN 14-12, Crane and Heavy Lift Issues Identified (IR 2414147)	
OE2014-01093	C&D Technologies, Inc. – Vendor Notification #49967	
OE2014-01138	C&D Technologies, Inc. (Part 21) – Log No. 2014-17-01	
OE2014-01231	C&D Technologies, Inc. (Part 21 Update) – Vendor Notification #49967	
OE2014-01287	C&D Technologies, Inc. (Part 21) – Log No. 2014-17-03	
PO 90164	RPS Power Supply Refurbishment	January 23, 2006
SER 91-322	Station Blackout Evaluation	
SQ-M-586	Drive Unit with Motor – Strainer	0
Technical Report No. 20-69-4333-E	Shutdown Heat Exchangers	January 9, 1970
TR-100248	EPRI Stationary Battery Guide: Design, Application, and Maintenance	2
TR-103834	EPRI Effects of Moisture on the Life of Power Plant Cables	August 1994

Miscellaneous

Number	Title	Revision/Date
TR-107044	EPRI Instrument Power Supply Tech Note	December 1996
TR-112175	EPRI Capacitor Application and Maint Guide	August 1999
TR-1001257	EPRI Capacitor Performance Monitoring Project	December 2000
WCAP-16755-NP	Operator Time Critical Action Program Standard	0
WIP file 019070	Seismic Analysis Of Raw Water Pumps	July 7, 1970

Procedures

Number	Title	Revision/Date
AOP-11	Loss of Component Cooling Water	16
AOP-17	Loss of instrument Air	15
AOP-18	Loss of Raw Water	8b
AOP-19	Loss of Shutdown Cooling	September 26, 2013
AOP-36	Loss of Spent Fuel Cooling	December 4, 2013
ARP-CB-10,11/A9	Annunciator Response Procedure A9 Control Room Annunciator A9	38
ARP-CB-20/A15	Annunciator Response Procedure Control Room Annunciator A15	42
ARP-CB-20/A17	Annunciator Response Procedure Control Room Annunciator A17	30
ARP-CB-20/A18	Annunciator Response Procedure Control Room Annunciator A18	29
CQE LIST Part One-Section II	Mechanical and Structural CQE List Requirements	42
EOP-00	Standard Post Trip Actions	31
EOP-07	Station Blackout	17
EOP/AOP	Attachments – MVA Vital Auxiliaries	0
EOP/AOP	Floating Steps	6
EM-CP-05-1B3B2	Calibration MCC-3B1 Feeder Breaker	7

Procedures

Number	Title	Revision/Date
EM-PM-EX-0200A	4160 Volt Circuit Breaker Inspection	25
EM-PM-EX-0200B	Spare 4160 Volt Circuit Breaker Inspection	2
EM-PM-EX-1400	4160 Volt Switchgear Inspection	40
EM-RR-EE-0900	Inspection and Maintenance for Station Battery No. 1 (EE-8A)	13a
EM-RR-EE-0901	Inspection and Maintenance for Station Battery No. 2 (EE-8A)	14
EM-RR-EE-0902	Emergency Load Profile Test for Station Battery No. 1 (EE-8A)	16
EM-RR-EE-0903	Emergency Load Profile Test for Station Battery No. 2 (EE-8B)	16
EM-ST-EE-0005	Capacity Discharge Test for Station Battery No. 1 (EE-8A)	20, 23, 25
EM-ST-EE-0006	Capacity Discharge Test for Station Battery No. 2 (EE-8B)	18, 21, 23
FCSG-56	Time Critical Operator Action Standard	3
GM-OI-HE-5	Intake Structure Overhead Crane Operation	9
MM-PM-AFW-0004	Fuel Oil Transfer Pump (FO-37) Maintenance	April 22, 2008
MM-RR-AE-0500	Missile Shield Removal	15
MM-RR-RC-0305	Removal of Reactor Vessel Closure Head, Hold Down Ring, and Upper Guide Structure	37
MM-RR-RC-0314	Reactor Vessel Closure Head Installation	28
NCM-1	Software Classification and Procurement	10
NPM-260	Shelf Life Instruction	8
NOD-QP-31	Operability Determination Process (ODP)	47
OI-AFW-1	Auxiliary Feedwater Actuation System Normal Operation	83
OI-AFW-4	Auxiliary Feedwater Startup and System Operation	October 24, 2013
OI-EE-2	480 Volt AC System Normal Operation	January 22, 2015
OI-SC-1	Shutdown Cooling Initiation	67

Procedures

Number	Title	Revision/Date
OPD-4-09	EOP/AOP Users Guidelines	20
OP-PM-AFW-0001	Auxiliary Feedwater System Flow Path Verification Using FW-6	13
OP-ST-AFW-3006	Auxiliary Feedwater System Category A and B Valve Exercise Test	7
OP-ST-AFW-3009	Auxiliary Feedwater Surveillance Test	27
OP-ST-AFW-3011	Auxiliary Feedwater Pump FW-10, Steam Isolation Valve, and Check Valve Tests	14
OP-ST-DG-0001	Surveillance Test EDG-1 Check	86
OP-ST-SHIFT-0001	Operations Technical Specification Required Shift Surveillance	120
OP-ST-SI-3002	Safety Injection System Category A, B and C Valve Exercise Test	October 8, 2013
OP-ST-VX-3002	Auxiliary Feedwater System Remote Position Indicator Verification Surveillance Test	7
PBD-30	Cables and Connectors	2
PED-CSS-12	Scaffold Construction	10
PED-EEI-7	Electrical Distribution System Load Control and Electrical Distribution System Load Log Application to Station Configuration Changes	4
PED-GEI-34	Evaluation of Seismically Induced System Interaction	19
PED-GEI-56	Configuration Change Closeout	32
PED-SEI-5	Methodology and Switch Setting Procedure	4
PED-SEI-10	Evaluation of MOV In-Situ Test Results	8
PED-SEI-28	Program Instructions	14
PLDBD-CS-52	Heavy Loads	22
SO-G-21	Modification Control	99
SO-O-23	Systems and Equipment Usage Data	February 12, 2015
SO-O-25	Temporary Modification Control	86
TBD-AOP-18	Loss of Raw Water	8a
TBD- AOP-11	Loss of Component Cooling Water	16
TBD-EOP-00	Standard Post Trip Actions	31

Procedures

Number	Title	Revision/Date
TBD-EOP-07	Station Blackout	17
TBD-EOP/AOP	Attachments – MVA Vital Auxiliaries	0
TBD-EOP/AOP	Floating Steps	6

Vendor Documents

Number	Title	Revision/Date
	Sigma Original Hand Pumps Double Acting Semi-Rotary Hand Wing Pumps Type “K” Standard Model – Cast Iron Body Bronze Model	
	Shutdown Heat Exchanger Specification Sheet	March 9, 1970
16577-C-0 (Q)	Civil And Structural Design Criteria For Standardized Nuclear Unit Power Plant System (SNUPPS) Wolf Creek Unit Only	November 18, 1988
AL-100	Sprague Aluminum Capacitors	1990
C&D Technologies Part 21 Notification	Final Report 10CFR Part 21 Evaluation Regarding Misaligned Separators in LCR-25 Standby Batteries	September 16, 2014
IM-09313133-1	NLI Operation and Maint. Manual for Replacement Diesel Generator Excitation/Voltage Regulator	4
Paladin DesignBase	Battery and Charger Sizing	December 2009
TD A038.0100	Acopian Single Output Power Supply	0
TD A038.0110	Acopian Dual Output Power Supply	0
TD C173.0020	Installation and Operating Instructions for C&D Flooded Cell Standby Battery	5
TD C173.0030	Specifications for C&D LCR Lead Calcium Standby Batteries	3
TD D088.0010	General Applications, Installation, Operation, Maintenance and Troubleshooting of Fuel Pumps	2
TD D088.0020	Instruction Manual for Series 3E Rotor Sizes 87, 87P and 95	1
TD D142.0110	Devar Delta T Power Reference	2
TD E982.0030	EGS NPE-525 52.5 VDC Power Supply	0
TD G080.4960	GE Molded Case Circuit Breakers	0

Vendor Documents

Number	Title	Revision/Date
TD L045.0020	Lambda LCS-A Series Power Supply	2
TD V125.0100	Technical Service Manual for Viking Heavy Duty Pumps Series 4194 & 495 Sizes GG-AL	1
TD W180.0030	Instruction Manual for Shutdown Heat Exchangers	1
TD W302.0030	Operating and Maintenance Instructions for 14 feet 0 inches I.D. Equipment Hatch	0
TVA 1347	Vishay (Sprague) TVA 1347 Data Sheet	August 12, 2011

Work Orders (WOs)

00125468	00361170	00422957	00440649	00520706
00265982	00361177	00439434	00440650	00522259
00265985	00365983	00439435	00440651	00522275
00303653	00367956	00439438	00440652	00522850
00313279	00367957	00439440	00440653	00522857
00314175	00370326	00440640	00440654	00523085
00320359	00370330	00440641	00440655	00526652
00337097	00377432	00440642	00469691	00529969
00340946	00387633	00440643	00470237	00534008
00343984	00388861	00440644	00489279	00914059
00343985	00393997	00440645	00496215	00920091
00359113	00393998	00440646	00469748	00950781
00359487	00396226	00440647	00502897	
00361133	00396227	00440648	00515634	

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Sincerely,

/RA/

Thomas R. Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000285/2015007
w/Attachment: Supplemental Information

Electronic Distribution for Fort Calhoun Station

Distribution
See next page

ADAMS Accession Number: ML15106A891

☒ SUNSI Review by: JLD	ADAMS: ☒ Yes ☐ No	☒ Non-Sensitive ☐ Sensitive	☒ Publicly Available ☐ Non-Publicly Available	Keyword: NRC-002

OFFICE	R3:RI/DRS	RI:DRS/EB1	RI:DRS/EB2	RI:DRS/EB1	SRI:DRS/EB1	C:DRP/D	C:DRS/EB1
NAME	JBozga	JBraisted	BCorrell	CSmith	JDixon	MHay	TFarnholtz
SIGNATURE	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/
DATE	4/2/15	4/6/15	4/2/15	4/2/15	4/8/15	4/14/15	4/16/15

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Letter to Louis Cortopassi from Thomas Farnholtz, dated April 16, 2015

SUBJECT: FORT CALHOUN – NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000285/2015007

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