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

May 14, 2014

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

Subject: FORT CALHOUN - NRC INTEGRATED INSPECTION REPORT NUMBER 05000285/2014007

Dear Mr. Cortopassi:

On March 31, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. On April 15, 2014, the NRC inspectors discussed the results of this inspection with Mr. Michael Prospero, Plant Manager, and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

During this inspection, the NRC staff examined activities conducted under your license as they relate to public health and safety with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC identified six findings that were evaluated under the risk significance determination process as having very low safety significance (green). The NRC determined five of these findings involved violations of NRC requirements. These violations are being treated as Non-cited Violations (NCVs), consistent with Section 2.3.2 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* (10 CFR) 2.390, "Public Inspections, Exemptions, Requests for Withholding," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000285/2014007
w/ Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2014007
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	February 16 through March 31, 2014
Inspectors:	J. Kirkland, Senior Resident Inspector J. Wingebach, Resident Inspector W. Smith, Project Engineer J. Bozga, Reactor Inspector
Approved By:	Michael Hay, Chief, Project Branch F Division of Reactor Projects

SUMMARY

IR 05000285/2014007; 02/16/2014 – 03/31/2014; Fort Calhoun Station; Integrated Resident Inspection Report and Confirmatory Action Letter Closeout Items.

The report covered a six-week period of inspection by the resident and regional inspectors from the NRC's Region III, and IV offices. Six findings of very low safety significance (Green) are 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, "Components 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 inspectors identified a Green finding for the licensee's failure to follow a procedure for classifying component failures. Specifically, the licensee's failure to follow Procedure FCSG-69-5, "Failure Identification and Reporting," is a performance deficiency. As a result, the failure of the Turbine-Driven Auxiliary Feedwater Pump, FW-10, to start on demand was not identified as a functional failure. Subsequently, the licensee properly evaluated the system performance taking into consideration the functional failure. The licensee documented the finding in the corrective action program as Condition Report 2014-04217.

The performance deficiency is more than minor, and therefore a finding, because if left uncorrected the performance deficiency could have the potential to lead to a more significant safety concern. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process For Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012. The finding is of very low safety significance (Green) because it did not affect the design or qualification of a mitigating system, structure, or component (SSC), represent a loss of system function, or loss of function of single or multiple trains of equipment. The finding had a human performance cross-cutting aspect associated with training because the licensee failed to provide adequate training to the engineering staff [H.9] (Section 1R12).

Green. The inspectors identified a green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to follow an operability determination procedure. Upon identifying that that a relief valve had not been tested within the required frequency the licensee failed to adequately address how this deficiency could affect the safety function of the component. Specifically, the licensee concluded the valve was operable based only on the consideration that it was not leaking. Subsequently, the licensee performed an evaluation providing adequate reasonable assurance of operability. The licensee documented the finding in the corrective action program as Condition Report 2014-03055.

The performance deficiency is more than minor, and therefore a finding, because if left uncorrected the failure to determine the ability of a structure, system, or component to perform its current licensing basis function in accordance with station procedures could lead to a more significant safety concern. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) For Findings

At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012, and determined that the finding is of very low safety significance (Green) because it did not affect the design or qualification of a mitigating SSC, represent a loss of system function or loss of function of single or multiple trains of equipment. The finding has a cross-cutting aspect in the human performance area because the licensee did not create and maintain complete, accurate, and up-to-date documentation [H.7] (Section 1R15).

Green. The inspectors identified a green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," involving the failure to establish and implement an adequate procedure for Post Maintenance Testing (PMT). Specifically, following maintenance on a raw water strainer the licensee's PMT failed to verify the flow capacity through the system required to determine operability. The failure to establish an adequate procedure to determine PMT is a performance deficiency. Subsequently, the licensee performed an adequate PMT verifying system flows were adequate and documented the deficiency in the corrective action program as Condition Report 2014-03084.

The performance deficiency is more-than-minor and therefore a finding because inadequate PMT following maintenance activities could adversely affect the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process For Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012, and determined that the finding is of very low safety significance (Green) because the finding did not affect the design or qualification of a mitigating SSC, represent a loss of system function or loss of function of single or multiple trains of equipment. The finding has a cross-cutting aspect in the area of problem identification and resolution because the licensee did not thoroughly evaluate issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance [P.2] (Section 1R19).

Green. The inspectors identified a green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to perform an operability determination as required by NOD-QP-31, "Operability Determinations Process (ODP)." Specifically, following the failure of an auxiliary building ventilation damper to open the licensee failed to evaluate the operability of equipment potentially impacted. Subsequently, the licensee performed an evaluation that provided reasonable assurance of operability. The licensee documented the finding in the corrective action program as Condition Report 2014-00211.

The performance deficiency is more than minor, and therefore a finding, because if left uncorrected the failure to determine the ability of a structure, system, or component to perform its current licensing basis function in accordance with station procedures could lead to a more significant safety concern. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) For Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012, and determined that the finding is of very low safety significance (Green) because the finding did not affect the design or qualification of a mitigating SSC, represent a loss of system function or loss of function of single or multiple trains of equipment. The finding has a cross-cutting aspect in the area human performance because the licensee did not provide training and ensure knowledge transfer to maintain a knowledgeable, technically competent workforce and instill nuclear safety values [H.9] (Section 4OA2).

Green. The inspectors identified multiple examples of a green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to promptly identify and correct conditions adverse to quality. Specifically, the licensee failed to take appropriate corrective action since 1997 when it was identified that the containment internal structure and auxiliary building had discrepant documentation between the size of structural beams and columns shown in drawings versus calculations. Subsequently, the licensee evaluated the non-conformances to provide a reasonable assurance of operability, and planned corrective actions to restore the structures to design basis requirements. The failure to correct conditions adverse to quality is a performance deficiency. The licensee documented the finding in the corrective action program as Condition Report 2014-04219.

The performance deficiency was determined to be more than minor because it adversely affected the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of the safety injection system and the shutdown cooling system. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) For Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012, and determined that the finding is of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating SSC that did not affect operability or functionality. The finding does not have a cross-cutting aspect because it is not reflective of current plant performance (Section 4OA4).

Green. The inspectors 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 design of the reactor vessel head stand met current licensing basis requirements. Specifically the design of the reactor vessel head stand did not meet the requirements as defined in the Updated Safety Analysis Report. Subsequently, the licensee evaluated the non-conformances to provide a reasonable assurance of operability, and planned corrective actions to restore the structures to design basis requirements. The failure to ensure the design of structures, systems, or components meet their current licensing basis is a performance deficiency. The licensee documented the finding in the corrective action program as Condition Report 2014-04218.

The performance deficiency was determined to be more than minor because it adversely affected the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of the safety injection system and the shutdown cooling system. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process (SDP)," Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs," dated May 25, 2004, and determined that the finding is of very low safety significance (Green) because the finding did not require quantitative assessment. The finding has a cross-cutting aspect in the area human performance because the licensee did not ensure the CIS at elevation 1045 ft. for storage of the reactor vessel head maintained adequate design margin [H.6] (Section 4OA4).

PLANT STATUS

The unit began the inspection period at 100% power. On March 17, 2014, the reactor manually scrammed due to a loss of stator water cooling to the generator. The unit was restarted on March 19, 2014, and reached 100% power on March 22, 2014, where it remained for the rest of the inspection period.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed partial system walk-downs of the following risk-significant systems:

- Partial system walk-down of the Raw Water System while component cooling water Heat Exchanger AC-1D was out of service on March 3, 2014, and

- Partial alignment of the turbine plant cooling water system following the restoration of the turbine plant cooling water heat exchanger on March 26, 2014.

The inspectors reviewed the licensee's procedures and system design information to determine the correct lineup for the systems. They visually verified that critical portions of the systems were correctly aligned for the existing plant configuration.

These activities constituted two partial system walk-down samples as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

.2 Complete Walk-down

a. Inspection Scope

On March 27, 2014, the inspectors performed a complete system walk-down inspection of the component cooling water system. The inspectors reviewed the licensee's procedures and system design information to determine the correct component cooling water system lineup for the existing plant configuration. The inspectors also reviewed open condition reports, in-process design changes, and other open items tracked by the licensee's operations and engineering departments. The inspectors visually verified that the system was correctly aligned for the existing plant configuration.

These activities constituted one complete system walk-down sample, as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors evaluated the licensee's fire protection program for operational status and material condition. The inspectors focused their inspection on the following two plant areas important to safety:

- March 27, 2014, 35A – Diesel Generator Room
- March 27, 2014, 35B – Diesel Generator Room

For each area, the inspectors evaluated the fire plan against defined hazards and defense-in-depth features in the licensee's fire protection program. The inspectors evaluated control of transient combustibles and ignition sources, fire detection and suppression systems, manual firefighting equipment and capability, passive fire protection features, and compensatory measures for degraded conditions.

These activities constituted two quarterly inspection samples, as defined in Inspection Procedure 71111.05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

.1 Review of Licensed Operator Performance

a. Inspection Scope

On March 14, 2014, the inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity due to main turbine testing.

The inspectors assessed the operators' adherence to plant procedures, including the conduct of operations procedure and other operations department policies.

These activities constitute completion of one quarterly licensed operator performance sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed two instances of degraded performance or condition of safety related structures, systems, and components.

- March 11, 2014, auxiliary feedwater Pump FW-10 steam admission Valve YCV-1045, failure to open

- March 28, 2014, stator water cooling system as a result of a conductivity probe leak

The inspectors reviewed the extent of condition of possible common cause failures of structures, systems, and components and evaluated the adequacy of the licensee's corrective actions. The inspectors reviewed the licensee's work practices to evaluate whether these may have played a role in the degradation of the structures, systems, and components. The inspectors assessed the licensee's characterization of the degradation in accordance with 10 CFR 50.65 (the Maintenance Rule), and verified that the licensee was appropriately tracking degraded performance and conditions in accordance with the Maintenance Rule.

These activities constituted completion of two maintenance effectiveness samples, as defined in Inspection Procedure 71111.12.

b. Findings

Introduction. The inspectors identified a Green finding for the licensee's failure to follow a procedure for classifying component failures. Specifically, following the failure of the turbine driven auxiliary feedwater Pump FW-10, the licensee failed to identify this as a functional failure in accordance with Procedure FCSG-69-5, "Failure Identification and Reporting."

Description. On February 12, 2014, the licensee was performing surveillance test IC-ST-IA-3009, "Operability Test of IA-YCV-1045-C and Close Stroke Test of YCV-1045." YCV-1045 is the steam admission valve for FW-10, and IA-YCV-1045-C is a check valve in the air system that can effect the position of YCV-1045.

During the test, YCV-1045 was given an open signal. The valve did not open as expected. Troubleshooting determined that a solenoid valve, YCV-1045-20-2 was not functioning properly. This solenoid valve would normally energize if FW-10 experienced a high discharge pressure event and close YCV-1045. After the high pressure trip would clear, the solenoid would de-energize to allow normal operation of YCV-1045. The solenoid valve properly changed state when energized but did not return to its original state when de-energized. As a result of the pumps failure to start, the station remained in a 24-hour shutdown action statement for approximately 19.5 hours, nearly 16 hours longer than planned.

After replacement, the solenoid valve was sent to a vendor for failure analysis. Failure analysis found metal shavings in the body of the valve, some of which were lodged between the plunger in the valve and the valve body causing it to stick in the energized position against spring pressure.

The licensee performed a maintenance rule functional failure evaluation and determined that no functional failure existed. The reasoning behind the determination was that the licensee believed the solenoid valve's function was to close YCV-1045 in the event of a high pressure trip signal, and that this did not affect the ability of the pump to start.

The inspectors questioned this determination; specifically why the function of the solenoid valve was evaluated instead of the function of the steam inlet valve, YCV-1045.

The inspectors noted that procedure FCSG-69-5, "Failure Identification and Reporting," which was issued to the station on December 17, 2013, states, in part, "any component failure that results in unplanned capability loss is always a functional failure." The station agreed that it did meet the definition of a functional failure by the procedure. The engineer had not referenced that procedure, was unaware of its existence, and had not been trained on it.

Analysis. The failure to follow procedure FCSG-69-5, "Failure Identification and Reporting," to determine if a functional failure of YCV-1045 occurred is a performance deficiency. As a result, the failure of FW-10 to start was incorrectly determined to not be a functional failure. The finding was more than minor because if left uncorrected the performance deficiency could have the potential to lead to a more significant safety concern. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) For Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012. They determined that the finding is of very low safety significance (Green) because the finding did not affect the design or qualification of a mitigating SSC, represent a loss of system function, or loss of function of single or multiple trains of equipment. The finding had a human performance cross-cutting aspect associated with training because the licensee failed to provide adequate training to the engineering staff [H.9].

Enforcement. This finding does not involve enforcement action because no violation of a regulatory requirement was identified. The licensee documented the finding in the corrective action program as Condition Report 2014-04217. Because the finding does not involve a violation and is of very low safety significance, it is being characterized as a finding FIN 05000285/2014-007-01, "Failure to Follow Procedures for Classifying Component Failures."

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed two risk assessments performed by the licensee prior to changes in plant configuration and the risk management actions taken by the licensee in response to elevated risk:

- March 27, 2014, Risk management actions associated with the west raw water header being out of service for maintenance

- March 17, 2014, Risk management actions associated with stator water cooling work that resulted in a turbine trip

The inspectors verified that these risk assessments were performed timely and in accordance with the requirements of 10 CFR 50.65 (the Maintenance Rule) and plant procedures. The inspectors reviewed the accuracy and completeness of the licensee's risk assessments and verified that the licensee implemented appropriate risk management actions based on the result of the assessments.

These activities constitute completion of two maintenance risk assessments and emergent work control inspection samples, as defined in Inspection Procedure 71111.13.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed one operability determination that the licensee performed for degraded or nonconforming structures, systems, or components.

- March 7, 2014, operability determination of thermal relief Valves SI-188 and SI-190

The inspectors reviewed the timeliness and technical adequacy of the licensee's evaluations. Where the licensee determined the degraded structures, systems, and components to be operable, the inspectors verified that the licensee's compensatory measures were appropriate to provide reasonable assurance of operability. The inspectors verified that the licensee had considered the effect of other degraded conditions on the operability of the degraded structures, systems, and components.

These activities constitute completion of one operability review sample, as defined in Inspection Procedure 71111.15.

b. Findings

i. Introduction. The inspectors identified a green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to follow an operability determination procedure. Specifically, the licensee's basis for operability did not adequately address the ability of the valve to provide overpressure protection.

ii. Description. On March 4, 2014, the licensee determined that thermal relief Valve SI-190, located in the high pressure safety injection system (HPSI), had not been tested within the specified frequency per ASME Code requirements. SI-190 is required to be tested every 48 months. A review of maintenance activities indicated that this valve was last tested on November 12, 2009 therfore, testing should have been completed before November 12, 2013.

SI-190 provides overpressure protection for HPSI piping and discharges to the pressurizer quench tank. SI-190 is also required to support the integrity of the HPSI system by not opening prior to its relief set point. The licensee's basis for operability was that they did not have indication of leakage rather than specifically evaluating against the specified safety functions.

The inspectors reviewed the immediate operability determination (IOD) and determined the licensee had not provided reasonable assurance that the valve was operable given it had not been tested as required per procedure. Specifically, the operability evaluation did not evalute the ability of the valve to provide overpressure protection and not relieve pressure to low or high. The inspecors noted the licensee inappropriately based operability on the condition that the valve was not currently leaking.

Procedure OP-FC-108-115-1002, "Supplemental Consideration for On-Shift Immediate Operability Determinations," requires the on-shift Senior Reactor Operator to identify the affected structures, systems, and components, review the licensing basis documents to identify the structures, systems, and components required function(s), identify how the affected part, component, subsystem, etc. contributes to the overall function of the structures, systems, and components, evaluate the effects of the condition and possible failure modes on the ability of the structures, systems, and components to perform its required functions, and document a clear statement as to whether the structures, systems, and components meets the conditions for being Operable or Inoperable.

Contrary to these requirements the licensee did not discuss the relief valve drifting low or relieving early; the ability of the valve to provide overpressure protection; or the effect if the valve drifted to the shut-off head of the Emergency Core Cooling pumps.

Based on the inspectors questions, the licensee supplemented the initial IOD to include all information required by IC-FC-108-115, "Operability Determinations."

The inspectors noted the licensee had difficulty determining the specified functions of SI-190. The inspectors reviewed the licensee's relief valve program basis document (PBD-9) that contains the basis for relief valve testing and functions. The inspectors noted this document had only been partially updated during the 2008 refueling outage.

<u>Analysis</u>. The failure to follow procedure IC-FC-108-115 is a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because if left uncorrected the failure to determine the ability of a structure, system, or component to perform its current licensing basis function in accordance with station procedures could lead to a more significant safety concern. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) For Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012, and determined that the finding is of very low safety significance (Green) because the finding did not affect the design or qualification of a mitigating SSC, represent a loss of system function or loss of function of single or multiple trains of equipment. The finding has

a cross-cutting aspect in the human performance area because the licensee did not create and maintain complete, accurate, and up-to-date documentation [H.7].

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" requires, 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 this, the licensee did not identify how the affected part, component, subsystem, etc. contributes to the overall function of the structures, systems, and components, evaluate the effects of the condition and possible failure modes on the ability of the SSC to perform its required function, and document a clear statement as to whether the structures, systems, and components meets the conditions for being operable or inoperable. Because the finding is of very low safety significance (Green) and has been entered into the corrective action program as CR 2014-03055, this violation is being treated as a non-cited violation, consistent with Section 2.3.2a of the NRC Enforcement Policy: NCV 05000285/2014007-02, "Failure To Follow An Immediate Operability Determination Procedure."

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed three post-maintenance testing activities that affected risk-significant structures, systems, or components.

- Post-maintenance testing following repairs of stator water cooling system on March 17, 2014;
- Post-maintenance testing following replacement of pressurizer pressure Indicator A/PIA-102Y on February 20, 2014; and
- Post-maintenance testing following raw water strainer maintenance on March 10, 2014.

The inspectors reviewed licensing and design-basis documents for the structures, systems, and components and the maintenance and post-maintenance test procedures. The inspectors observed the performance of the post-maintenance tests to verify that the licensee performed the tests in accordance with approved procedures, satisfied the established acceptance criteria, and restored the operability of the affected structures, systems, and components.

These activities constitute completion of three post-maintenance testing inspection samples, as defined in Inspection Procedure 71111.19.

b. Findings

Introduction. The inspectors identified a green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to implement an adequate procedure for Post Maintenance Testing (PMT) following maintenance activities on a raw water strainer.

Description. On March 10, 2014, the inspectors reviewed the PMT that the licensee had scheduled to perform following maintenance on a raw water strainer. The inspectors questioned the listed acceptance criteria of differential pressure. The function of the strainer is to provide strained water to heat exchangers. If the strainters start to foul then flow would decrease. Differential pressure would be an indication of fouling however this value would then need to be translated to flow and checked against flow amounts required for operability. The inspector noted this procedure did not translate the differential pressure to a total system flow and therefore it did not establish operability of the raw water system.

Upon questioning by the inspectors the licensee determined the listed PMT was not adequate to determine operability. The licensee changed the PMT to a different procedure that verifies adequate strainer flow.

The Purpose of PMT is to provide assurance that a component and its associated subsystems are functional after completion of maintenance. The level of PMT performed should be appropriate for the scope of maintenance work performed and the required design functions of the component. Completion of PMT provides a high degree of confidence that a component is capable of performing its design functions.

The inspectors noted the following condition reports in the corrective action program regarding concerns with the PMT program. The following condition reports are examples:

- CR 2014-03192: Documents an increase in CRs documenting inadequate PMTs in the preventative maintenance program shows an emerging trend.

- CR 2014-02360: Documents a trend identified that the planning for maintenance activites does not always identify the correct and necessary PMT.

- CR 2014-02363: Documents a trend identified that PMT is not always properly tracked or scheduled.

The NRC plans to perform follow up inspections the determine the effectiveness of licensee actions to resolve these adverse trends related to the PMT program.

Analysis. The failure to implement an adequate procedure for PMT is a performance deficiency. The performance deficiency is more than minor and therefore a finding because if left uncorrected the performance deficiency has the potential to lead to a more significant safety concern. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) For Findings At-Power,” Exhibit 2, “Mitigating Systems Screening Questions,” dated June 19, 2012, and determined that the finding is of very low safety significance (Green) because the finding did not affect the design or qualification of a mitigating SSC, represent a loss of system function or loss of function of single or multiple trains of equipment. The finding has a cross-cutting aspect in the area of problem identification and resolution because the licensee did not thoroughly evaluate issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance [P.2].

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" requires, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawing." Contrary to this, the licensee procedure for PMT was not appropriate in that it failed to ensure the appropriate acceptance criteria was established to verify operability. Because the finding is of very low safety significance (Green) and has been entered into the corrective action program as CR 2014-03084, this violation is being treated as a non-cited violation, consistent with Section 2.3.2a of the NRC Enforcement Policy: NCV 05000285/2014007-03, "Failure to establish an adequate PMT procedure."

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures (MS05)

a. Inspection Scope

For the period of January 1, 2013, through December 31, 2013, the inspectors reviewed licensee event reports (LERs), maintenance rule evaluations, and other records that could indicate whether safety system functional failures had occurred. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, and NUREG-1022, "Event Reporting Guidelines: 10 CFR Part 50.72 and 50.73," Revision 3, to determine the accuracy of the data reported.

These activities constituted verification of the safety system functional failures performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review

a. Inspection Scope

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

b. Findings

No findings were identified.

.2 Semiannual Trend Review

a. Inspection Scope

The inspectors reviewed the licensee's corrective action program, performance indicators, system health reports, corrective action program trend reports, Root Cause Analysis 2013-08675 "Ineffectiveness in the FCS Corrective Action Program," Nuclear Oversight performance assessments, and other documentation to identify trends that might indicate the existence of a more significant safety issue. The inspectors verified that the licensee was taking corrective actions to address identified adverse trends.

b. Observations

The inspectors identified an adverse trend with respect to the licensees ability to identify and correct adverse trends. In particular, the inspectors noted during review of condition reports that many were not coded and tracked as procedural use and adherence concerns when cleary the subject in the condition report was the result of procedural use and adherence deficiencies. In addition, the inspectors reviewed corrective actions for trends that the licensee identified as adverse. The inspectors reviewed the licensee's corrective action procedures to determine what the station expectation for corrective action when an adverse trend is identified. The inspectors were not able to follow a prescriptive process to consistently determine what actions should be taken when an adverse trend is found. Specifically it was generally up to the condition report screening committee what level and severity the condition report was assigned and who the condition report was assigned to for follow up. The inspectors determined that since the level and severity directly affect the effort of investigation the licensee might not consistently respond to adverse trends.

These activities constitute completion of one semiannual trend review sample, as defined in Inspection Procedure 71152.

b. Findings

No findings were identified

.3 Annual Follow-up of Selected Issues

a. Inspection Scope

The inspectors selected one issue for an in-depth follow-up:

- Ventilation exhaust Damper HCV-825B failed to open on November 15, 2013

  The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews, and compensatory actions. The inspectors verified that the licensee appropriately prioritized the planned corrective actions and that these actions were adequate to correct the condition.

These activities constitute completion of one annual follow-up sample, as defined in Inspection Procedure 71152.

b. Findings

Introduction. The inspectors identified a green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to perform an immediate operability determination (IOD) as required by NOD-QP-31, "Operability Determinations Process."

Description. On November 15, 2013, auxiliary building ventilation exhaust Damper HCV-825B failed to open as required upon the start of exhaust Fan VA-40B. Condition Report 2013-21373 was initiated. The auxiliary building ventilation system is designed to control the temperature of safety related equipment and instrumentation and controls outside the control room but inside the auxiliary building and filter radioactive material to maintain doses within regulatory limits.

The inspectors reviewed the IOD associated with CR 2013-21373 and determined the licensee failed to evaluate the effect of the degraded condition on the ability of the auxiliary building ventiliation system to perform its support functions as described above. NOD-QP-31, section 1.2.2, "Scope of structures, systems, and components for Operability Determinations" states, in part, "structures, systems, and components that are not explicitly required to be operable by TSs, but perform required support functions (as specified by the TSs definition of operability) for SSCs that are required to be operable by TSs." The licensee determined that HCV-825B was non-functional and closed the condition report to a work request. The station did not assess the ability of VA-40B to perform its function, the overall ability of the auxiliary building controlled area ventilation system to perform its current licensing basis function, or the support function for other components that are required by technical specifications (TS) such as to provide room cooling for the electrical distribution system and the emergency core cooling pump motors. Additionally, during this timeframe auxiliary building ventilation system exhaust Fan VA-40C was also non-functional.

Based on the inspectors questions the licensee initiated CR 2014-00203 to address the inspectors concerns. The licensee appropriately evaluated the auxiliary building ventilation system's ability to perform its function while in a degraded condition and determined the system was operable. The licensee initiated CR 2014-00211 which acknowledged a misunderstanding regarding all the support functions of the auxiliary building ventilation system that were not properly evaluated.

Analysis. The failure to perform an immediate operability determination as required by NOD-QP-31, "Operability Determinations Process " is a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because if left uncorrected the failure to determine the ability of a structure, system, or component to perform its current licensing basis function in accordance with station procedures could lead to a more significant safety concern. The inspectors evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) For Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012, and determined that the finding is of very low safety significance

(Green) because the finding did not affect the design or qualification of a mitigating SSC, represent a loss of system function or loss of function of single or multiple trains of equipment. The finding has a cross-cutting aspect in the area human performance because the licensee did not provide training and ensure knowledge transfer to maintain a knowledgeable and technically competent workforce and instill nuclear safety values [H.9].

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings” requires, in part, “Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawing.” Contrary to this, the licensee did not perform an immediate operability determination as required by NOD-QP-31, “Operability Determinations Process (ODP).” Because the finding is of very low safety significance (Green) and has been entered into the corrective action program as CR 2014-00211, this violation is being treated as a non-cited violation, consistent with Section 2.3.2a of the NRC Enforcement Policy: NCV 05000285/2014007-04, “Failure To Perform An Immediate Operability Determination.”

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

.1 (Opened and Closed) Licensee Event Report 05000285/2014-001-00: Reactor Shutdown due to Sluice Gate Failure

At approximately 10:30 p.m. on January 8, 2014, traveling screen sluice Gate CW-14C motor operator shaft was found by Operations personnel to be damaged (bent). One hour later a large block of ice buildup was observed on top of the sluice gate caused by a pinhole leak in the backwash piping located directly above the CW-14C gate and freezing temperatures. At 2:50 a.m. on January 9, 2014, Operations attempted to manual close the sluice gate with no success. At 3:15 a.m. the station entered TS 2.0.1(1) due to all raw water (RW) pumps being declared inoperable and at 5:18 a.m. the station commenced a reactor shutdown. At 9:00 a.m. the station completed the reactor shutdown.

The licensee determined the sluice gate motor operator shaft bent because the motor torque setting was set to high. Sluice Gate CW-14C was uncoupled from the shaft, lowered by chainfall, and then verified closed by divers. At 3:50 a.m., on January 10, 2014, all raw water pumps were declared operable and TS 2.0.1(1) was exited.

The inspectors reviewed the causal analysis for the improper sluice gate motor torque setting. The licensee identified that the sluice gates had not been previously credited as safe shutdown equipment. Even though the sluice gates recently became credited as safe shutdown equipment the station failed to enter the sluice gate motor operated valves into the MOV program, perform additional analyses, and implement necessary preventative maintenance (PM) activities.

The inspectors had previously issued a violation to the licensee for failure to classify the sluice gates and MOVs as safety class III components (violation 05000285/2012002-02). This licensee event report is closed.

.2 <u>(Opened and Closed) Licensee Event Report 05000285/2014-002-00:</u> Reactor Manual Trip due to Control Rod Misalignment

After reaching criticality on January 12, 2014, the control room attempted to reduce power ascension rate while at zero percent power by inserting Group 4 control element assemblies (CEAs). All Group 4 CEAs inserted with the exception of control Rod RC-10-41, which failed to move. Group 4 was inserted further until a 10 inch deviation existed between RC-10-41 and the remaining Group 4 CEAs. Power continued to slowly rise and the operators conservatively decided to manually trip the reactor.

Troubleshooting determined that the rectifier for RC-10-41 had failed. The failed control element drive mechanism rectifier and associated fuses for RC-10-41 were replaced.

The inspectors reviewed the licensee's causal analysis that determined the rectifier had failed. The cause of the failure was not age related, thus it was not reasonable to assume that the licensee could have foreseen the failure. This licensee event report is closed.

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

4OA4 IMC 0350 Inspection Activities (92702)

Inspectors continued implementing IMC 0350 oversigt activities, which include documentation of the inspections results associated with the restart checklist items contained in the Confirmatory Action Letter (CAL) issued February 26, 2013 (EA-13-020, ML 13057A287). The purpose of these inspection activities was to assess the licensee's performance and progress in addressing its implementation and effectiveness of FCS's Integrated Performance Improvement Plan (IPIP), significant performance issues, weaknesses in programs and processes, and flood restoration activities.

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

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

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

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

(1) Item 2.d: Containment Internal Structure

i. Background

In the letter that closed Confirmatory Action Letter CAL-13-20 (ADAMS ML13351A423), the NRC stated that:

> Item 2.d was required to be completed prior to restart. The NRC determined that OPPD appropriately evaluated the cause and extent-of-condition for this issue and independently verified that OPPD has actions in place via the corrective action process to restore the containment internal structure to its design criteria in a timely manner. Following completion of NRC inspection activities, on November 7, 2013, the IMC 0350 Panel and NRC staff involved in reviewing Restart Checklist Item 2.d conducted discussions and determined this item had been adequately addressed by OPPD and therefore closed it. The inspection activities associated with this this item will be documented in NRC Inspection Report 05000285/2013-013.

However, because of delays, the documentation of the inspection activities associated with this item are closed in this report.

The containment internal structure (CIS) at Fort Calhoun Station (FCS) is an independent, multi-story, reinforced concrete structure located inside the containment shell. The CIS is comprised of 269 structural elements (135 reinforced concrete beams, 32 reinforced concrete column sections, and 102 reinforced concrete slabs) that physically support safety related components – including the reactor coolant system, emergency core cooling system piping, and the reactor components.

On May 22, 2012, the licensee wrote a condition report to document non-conforming conditions for several structural elements of the CIS on floor elevations 1013 ft., 1045 ft., and 1060 ft. Specifically, the licensee discovered numerous errors and discrepancies between the design drawings and design calculations. Errors included the failure to meet both working stress and ultimate strength design criteria as required by the building's construction code, multiple calculations of record, discrepancies between as-built design drawings and calculations, discrepancies in loading values for the same structures, unchecked load combinations, assumptions without justification, numerical errors, poor legibility, lacking calculations, etc.

The extent of condition for non-conforming structural calculations affected all Class 1 structures onsite because the errors were suspected to be introduced by the architecture engineering firm. However, the licensee had recently reconstituted the design basis information for the containment shell, containment dome, and the intake structure, so the only affected structure (besides the containment internal structure) was the auxiliary building.

ii. Inspection Scope

The inspectors reviewed the licensee's Root Cause Analysis (RCA), corrective actions, extent of condition, calculations, operability assessments, and Licensee Event Report (LER) 2012-014 associated with the non-conforming CIS.

iii. Methodology

The FCS design and licensing basis states that the design code of record for the CIS is the 1963 edition of American Concrete Institute Standard 318: Building Code Requirements for Reinforced Concrete (ACI 318-63). The Final Safety Analysis Report requires the CIS to meet the more restrictive of two ACI 318-63 design criteria: (1) working stress and (2) ultimate strength. However, for the purposes of the operability assessment, the capability of the CIS was compared to the ultimate strength design method because this method is specified by the FSAR to ensure no loss of function. Because the CIS drawings and calculations were not in agreement, the licensee performed a reconstitution of the CIS calculations. To accomplish this task, several state-of-the-art computer codes were employed.
First, the CIS geometry, properties, and loads were determined. For the computer model to provide accurate results, the CIS structural geometrical properties were developed by verifying the drawings were consistent with the as-built design by performing field walk-downs. The NRC independently verified that the walk-downs and the CIS structural geometry used in the analysis were adequate.

Next, the CIS material properties were established. The FSAR specified specific values for the CIS construction (compressive strength, rebar strength, density, modulus of elasticity, etc.), but the licensee opted to use plant-specific data taken from certified mill test reports and other material test data for the operability assessment. This technique required specific analytical methods in accordance with NRC accepted practices because these values were in excess of the FSAR specified values (i.e. the structure was physically stronger than the FSAR specified values.)

Finally, the CIS loads were developed. Similar to the CIS geometry field walk-downs, the licensee reviewed the self-weight of the superstructure, live floor loads, accident pressure loads (from GOTHIC computer analysis), seismic loads, equipment loads, piping loads, commodity loads, and outage loads. With the exception of the accident pressure loads, the loads were determined through drawings and physical walk-downs of the CIS. The NRC verified that the loads were consistent with the walk-downs and were adequate for the operability assessment.

For the accident loads, the licensee used the GOTHIC computer code. Due to the large scope and technical complexity of the issue, the NRC regional inspectors created a Task Interface Agreement (TIA) with NRC Headquarters experts for technical assistance in reviewing the CIS accident pressure loads. TIA 2013-03 (ADAMS ML14085A184) reviewed the licensee's GOTHIC thermal hydraulic analysis. This analysis used computational methods to determine the maximum pressure loads on the internal structure. NRC Headquarters technical reviewers evaluated the licensee's methodology and use of the computer code GOTHIC, the mass and energy release, assumptions (boundary conditions, flow paths, and input parameter sensitivity) in the model, and the overall results. TIA 2013-03 concluded that the pressure loads generated from the GOTHIC computer analysis was acceptable.

With the CIS geometry, material properties, and loads determined, the operability assessment used the computer code finite element software GTSTRUDL (Structural Design Language) to determine the total demand (moments, shears, and forces) on the CIS elements. The output of the GTSTRUDL was then compared to an automated analysis that determined the capacity of each structural element based on the ultimate strength design method of ACI 318-63.

iv. Assessment and Results

Root Cause Assessment

The inspectors reviewed the licensee's root cause analysis and determined it was adequately performed. However, the licensee concluded the root cause of this event was indeterminate due to the lack of documentation for review, lack of ability to interview personnel directly involved, and lack of knowledge of oversight process in 1968. The team noted that the probable causes identified by the licensee (lack of review of the design calculations, lack of oversight of a contractor, poor documentation, and lack of design basis information) were previously identified as root causes for poor station performance and the main contributors to the plant's performance decline and transition to Inspection Manual Chapter 0350. The team concluded that the most likely cause(s) were in alignment with the licensee's probable cause – inadequate vendor oversight coupled with poor recordkeeping practices and a lack of a comprehensive and clear design basis. These observations are aligned with previous NRC Inspection Manual Chapter 0350 Inspection Reports.

Corrective actions

The licensee developed a number of corrective actions to address the issues, which included:

- Performing a design bases reconstitution to identify and define the licensing and design bases to assure documentation remains current, accurate, complete, and retrievable – specifically for the containment internal structure and the auxiliary building, and

- Conduct training with engineering personnel to address proper use of design and licensing bases information.

The inspectors determined the licensee had adequately addressed corrective actions for the identified probable cause.

Extent of condition review

As stated in the background section, the affected structures were the CIS and the auxiliary building. The GTSTRUDL analyses of the auxiliary building identified that all of the structural members met the ultimate strength design method of ACI 318-63 and were therefore operable. The inspection team determined this was acceptable because the licensee recently re-constituted the design basis for the intake structure and for the containment shell in support of a reactor coolant system upgrade. However, the inspection team noted that the CIS discrepancies with calculations and drawings had been identified during a previous design basis reconstitution. The NRC identified a number of these concerns were found to be closed without sufficient corrective action. The “Design Basis Documents (DBD) Open Items Lists” present a very large potential extent of condition issue for latent engineering deficiencies at the station. Previous NRC inspection teams have also identified regulatory concerns related to the quality of the “Design Basis Documents”.

CIS Operability and Corrective Actions Review

The licensee’s GTSTRUDL analysis of the CIS demonstrated that 261 out of the 269 structural elements met the ultimate strength design method of ACI 318-63. The NRC chose to focus their review on the remaining 8 beams that exceeded the ACI code requirements.

To demonstrate operability, the licensee employed non-linear computer finite element analysis codes (ANSYS and SOLVIA) for the beams that exceeded the ultimate strength design criteria to show they possess adequate strength to perform their required function, though beyond design limits.

Four of the eight beams that exceeded ACI 318-63 support the safety injection tanks, and the remaining four support the reactor vessel head (RVH) laydown area that is used to store the reactor head during refueling outages.

The NRC Region IV Office in coordination with technical experts from Region III and NRR, reviewed the finite element analyses that form the basis for the operability determination of the CIS. TIA 2013-05 “Containment internal structures operability calculations at Fort Calhoun Station,” was developed to evaluate the technical adequacy of the licensee’s GTSTRUDL analysis and operability assessment. TIA 2013-05 (ADAMS ML14016A260), determined that the operability assessment used reasonable inputs to the GTSTRUDL models and found the conclusions were acceptable.

In addition, the TIA concluded that it was acceptable for the licensee to use a non-linear finite element analysis method, incorporating inelastic material response, to demonstrate functionality of those CIS elements that are not in compliance with the ACI 318-63.

Beam 22

For the 4 beams that support the safety injection tanks, it was determined that a set of beams, Beam-22a and Beam-22b, represented the worst-case. The NRC inspectors reviewed the ANSYS finite element calculation FC08235, “Capacity Calculation of B-22 Using Finite Element Analysis,” which calculated the ultimate shear capacity of beams B-22a and B-22b at floor elevation 1013 ft.

The outputs from the FC08235 calculation for beams B-22a and B-22b were subsequently utilized in calculation FC 08189 to demonstrate operability of these beams.

The licensee used the available test data documented in a research paper, Vecchio, Frank J, “On the Post-Peak Ductility of Shear-Critical Beams,” ACI Special Publication 237 (2006):109 to demonstrate that the structure would not fail even if it exceeded the ACI 318-63 code limits. Specifically, the paper discusses the finding that structures have margin beyond the calculated ultimate strength design of ACI 318-63.

The staff reviewed the academic paper by Vecchio and found that the paper used different inputs in the model than the licensee used in calculation FC08235. It is imperative to note that the functionality of B-22a and B-22b totally hinges on the accuracy and reasonableness of the predicted peak load derived from ANSYS calculation FC08235. Because there is little margin in the capacity of the beam (less than 8%), and considering the results of the sensitivity analysis of the concrete tensile strength, the inspectors identified a concern regarding the calculated safety margin for beams B-22a and B-22b during postulated design basis accident conditions.

Specifically, since calculation FC 08235 did not quantify the accuracy of results from the finite element model used to represent beams B-22a and B-22b, there was a concern that the uncertainty could adversely impact the safety margin. In response to the inspectors concerns, the licensee performed additional finite element analyses, FC 08314, “ANSYS Analysis of Benchmark Beams,” and demonstrated the accuracy of the finite element model by comparing the finite element model results against the test results documented in the research paper.

In order to increase the safety margin for beams Beam-22a and Beam-22b, the licensee reduced the load from the safety injection tank, which represents a significant load on these beams, by limiting the safety injection tank level from 100% to 74%. The licensee revised their operating procedure OI-SI-1, “Safety Injection – Normal Operations,” to incorporate this change. The NRC verified the compensatory measure to reduce the volume in the safety injection tanks have been established and are complied with. Additionally, the licensee committed[1] to resolve any deficiencies associated with beams B22a and B-22b in accordance with the corrective action program.

<u>Reactor Vessel Head Laydown Area</u>

The licensee provided the inspectors with technical reports and calculations that evaluated the structural adequacy of the northwest portion of the CIS at floor elevation 1045 ft. which supported the placement of the Reactor Vessel Head (RVH). The main calculation for RVH laydown area FC06971, “RV Head Laydown Area Seismic Analysis”, used a non-linear finite element code called SOLVIA.

The NRC technical review team provided extensive review of FC06971 and related calculations, and held multiple interactions with the licensee’s engineers and consultants over several months to discuss NRC concerns and questions. As a result, the licensee’s analysis and associated calculations were subject to several iterative changes during that timeframe.

The NRC technical review team concluded that the licensee provided reasonable assurance that the CIS had sufficient structural capacity to perform its design functions if subjected to a postulated design basis earthquake, differential pressure, equipment loads, live loads, and dead loads. Specifically,

- The methodology and inputs used in the structural evaluations were reasonable and appropriate,

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[1] Confirmatory Action Letter EA-13-243 (ADAMS ML13351A395)

- The finite element model adequately reflected the load path through the structure for the RVH head laydown condition, and

- The licensee calculations adequately evaluated design basis earthquake, differential pressure, and dead loads.

However, because of the low safety margin for the RVH structure, the licensee committed[2] to evaluate the structural design margin for the reactor head stand and resolve any deficiencies in accordance with the corrective action program prior to the next use of the reactor head stand.

This activity constitutes completion of Action Items 2.d.1, 2.d.2, 2.d.3 and LER 2012-014 as described in Confirmatory Action Letter 4-12-002. This restart checklist item is closed.

v. Findings

i. Introduction. The NRC inspection team identified multiple examples of a green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to promptly identify and correct conditions adverse to quality. Specifically, since 1997 the licensee identified that the containment internal structure and auxiliary building had discrepant documentation between the size of structural beams and columns shown in drawings versus calculations and failed to correct the deficiencies in a timely manner.

Description. During preparations for piping modifications in 2012, the licensee identified that several structural elements within the containment internal structure and auxiliary building had multiple discrepancies between the design drawings and the design calculations. As a result of the discrepancies, several of the beams and columns of the containment internal structure and auxiliary building were not built or designed according to the licensee's construction code of record, ACI 318-63, "Building Code Requirements for Reinforced Concrete."

Several NRC inspections at FCS, dating back to mid-1980's, documented weaknesses in the licensee's design control process. Corrective actions for these issues included a reconstitution of the design bases. NRC Inspection Report 50-285/2012-11 (ADAMS ML12366A158), provides an overview of the design bases reconstitution process and documented a non-cited violation for the station's failure to follow procedures that required periodic updates and maintenance of the design basis documents (DBDs).

During subsequent NRC inspections, the team identified that several unresolved, or "open items" from the licensee's design bases reconstitution project were closed without corrective actions or sufficient justification to delay corrective action implementation. Specifically, several open items from the licensee's

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[2] Confirmatory Action Letter EA-13-243 (ADAMS ML13351A395)

design bases document SDBD-CONT-501 identified the discrepancies with the columns and beams. However, in a memorandum dated February 13, 1997, the licensee closed the open items without correcting the conditions adverse to quality.

Specifically, the closure of Open Item #6 states, “This open item deals with discrepant documentation regarding beam sizes shown on Fort Calhoun drawings versus the design calculations.” Open Item #10 states “calculations could not be located for a number of structural elements within the containment…the safety injection tanks are placed directly on beams B-22 which carry a portion of the tank load… there is no specific analysis for this portion … the lack of design calculations for isolated reinforced concrete containment structural elements is acceptable. This open item is considered closed.”

The NRC inspection team identified that the DBD Open Items were closed inappropriately, without adequate corrective actions to resolve the deficiencies between the design drawings and the design calculations for a large population of beams and columns for the containment and auxiliary building. The effect of the design calculation and drawing discrepancies was that major portions of the containment internal structure and auxiliary building structure were not in compliance with the construction code of record, ACI 318-63.

Analysis. The inspectors determined the licensee’s failure to correct the deficiencies in the design drawings and calculations for the containment internal structure and auxiliary building beams and columns contrary to the corrective action measures requirements per 10 CFR Part 50, Appendix B, Criterion XVI and was a performance deficiency.

In accordance with Inspection Manual Chapter (IMC) 0612, “Issue Screening”, Appendix B, the inspectors determined the performance deficiency affected the Mitigating Systems Cornerstone. The performance deficiency was determined to be more than minor because it adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of the safety injection system and the shutdown cooling system.

The inspectors evaluated the finding using IMC 0609, Appendix A, “The Significance Determination Process (SDP) For Findings At-Power,” Exhibit 2, “Mitigating Systems Screening Questions,” dated June 19, 2012, and determined that the finding is of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating SSC that did not affect operability or functionality. The finding does not have a cross-cutting aspect because it is not reflective of current plant performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action” states, in part, “Measures shall be established to assure that conditions adverse to quality…are promptly identified and corrected.” Contrary to this requirement, from February 13, 1997 to present, the inspectors determined the licensee failed to correct conditions adverse to quality for the containment internal structure and auxiliary building design drawing and design calculation discrepancies. Specifically, the inappropriately closed Open Items for design calculations and design drawing discrepancies culminated in several beams and columns within the containment internal structure and auxiliary building not meeting the requirements of ACI 318-63. Because this violation was of very low safety significance (Green) and it was entered into the licensee's corrective action program as Condition Report 2014-04219, this violation is being treated as an NCV, consistent with Section 2.3.2a of the NRC Enforcement Policy (NCV 05000285/2014007-05), "Failure To Correct Conditions Adverse To Quality In The Containment Internal Structure And Auxiliary Building."

ii. Introduction. The NRC inspection 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 reactor vessel head stand structure was designed in accordance with current licensing basis requirements. Specifically the design did not meet the Updated Safety Analysis Report (USAR) requirements.

Description. The reactor vessel stand structure is located on elevation 1045 ft. of the containment internal structure, and is used for the storage of the reactor vessel head during outage conditions. Section 5.5.2.2 of the USAR provides structural design requirements for the containment internal structure. Specifically, the USAR requires that the containment internal structure is designed in accordance with the 1963 edition of the American Concrete Institute Standard 318: Building Code Requirements for Reinforced Concrete (ACI 318-63).

ACI 318-63 requires the working stress design method loading combination of dead load plus live load. Calculation FC 7176, "Assessment of Concrete Beams at Elev. 1045'-0" in Containment for Rx Vessel Head Load", Revision 2 evaluated the structural adequacy of the containment internal structure at elevation 1045 ft. for storage of the reactor vessel head during outage conditions. The inspectors reviewed this calculation and identified that the calculation used the ultimate strength design method in the 2002 edition of ACI 318. The design basis code of record is ACI 318-63, and requires the working stress design method. Compliance with ACI 318-63 requirements ensures the structure will remain elastic when subjected to the applied loads. Fort Calhoun Station does not have a license amendment approving the use of the 2002 edition of ACI 318 or the ultimate strength design method.

Analysis. The inspectors determined the licensee's failure ensure the reactor vessel head stand structure was designed in accordance with ACI 318-63, as required by the USAR, was a performance deficiency and contrary to the design control measures per 10 CFR Part 50, Appendix B, Criterion III.

In accordance with Inspection Manual Chapter (IMC) 0612, "Issue Screening", Appendix B, the inspectors determined the performance deficiency affected the Mitigating Systems Cornerstone. The performance deficiency was determined to be more than minor because it adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of the safety injection system and the shutdown cooling system.

The inspectors evaluated the finding using IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process (SDP)," Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Operational

Checklists for Both PWRs and BWRs," dated May 25, 2004, because the reactor vessel head stand area is only used during outage conditions when the reactor is shutdown.

The inspectors determined there was no impact on the safety functions of core heat removal, RCS inventory control, power availability, containment control, and reactivity control as required by Checklist 4, "PWR Refueling Operation: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer." Since there was no impact on these functions, the finding did not require quantitative assessment and is of very low safety significance (Green). The finding has a cross-cutting aspect in the area human performance because the licensee did not ensure design of the reactor vessel head structure maintained adequate design margin [H.6].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, "The 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 this requirement, since April 16, 2013, the inspectors determined the licensee failed to ensure the design of the reactor vessel head stand met current licensing basis requirements. Specifically, calculation FC 07176, Revision 2, did not demonstrate compliance with the working stress design method of ACI 318-63. Because this violation was of very low safety significance (Green) and it was entered into the licensee's corrective action program as Condition Report 2013-08499 and 2014-04218, this violation is being treated as an NCV, consistent with Section 2.3.2a of the NRC Enforcement Policy (NCV 05000285/2014007-06), "Failure To Check The Adequacy Of The Design For The Reactor Vessel Head Structural Elements."

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 15, 2014 the inspectors presented the inspection results to Mr. Michael Prospero, Plant Manager, 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.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Anderson, Manager, Design Engineering
D. Bakalar, Manager, Security
J. Bousum, Manager, Emergency Planning and Administration
C. Cameron, Supervisor Regulatory Compliance
L. Cortopassi, Site Vice President
M. Ferm, Manager, System Engineering
K. Ihnen, Manager, Site Nuclear Oversight
T. Lindsey, Director, Training
E. Matzke, Senior Licensing Engineer, Regulatory Assurance
J. McManus, Manager, Engineering Programs
B. Obermeyer, Manager, Corrective Action Program
M. Prospero, Plant Manager
T. Orth, Director, Site Work Management
S. Shea, Supervisor, Operations Training
T. Simpkin, Manager, Site Regulatory Assurance
S. Swanson, Director, Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000285/2014-007-01	FIN	Failure to Follow Procedures for Classifying Component Failures (Section 1R12)
05000285/2014-001-00	LER	Reactor Shutdown due to Sluice Gate Failure (Section 4OA3)
05000285/2014-002-00	LER	Reactor Manual Trip due to Control Rod Misalignment (Section 4OA3)

Opened and Closed

05000285/2014-007-02	NCV	Failure to follow an immediate operability determination procedure (Section 1R15)
05000285/2014-007-03	NCV	Failure to implement an adequate PMT procedure (Section 1R19)
05000285/2014-007-04	NCV	Failure to perform an immediate operability determination (Section 4OA2)
05000285/2014-007-05	NCV	Failure to correct conditions adverse to quality in the containment internal structure and auxiliary building (Section 4OA4)
05000285/2014-007-06	NCV	Failure to check the adequacy of the design for the Reactor Vessel Head structural elements (Section 4OA4)

Closed

05000285/2012-014-1	LER	Containment Beam 22 Loading Conditions Outside of the Allowable Limits

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures

Number	Title	Revision
OI-BW-1	Bearing Water System Normal Operation	27
OI-CC-1-CL-A	Component Cooling Checklist	77
OI-RW-1	Raw Water System Normal Operation	108

Drawings

Number	Title	Revision
11405-M-100-001	Flow Diagram Raw Water System P&ID	102
11405-M-258-COV	Composite Flow Diagram Turbine Plant Cooling Water System P&ID	13
11405-M-258-001	Flow Diagram Turbine Plant Cooling Water System P&ID	45
11405-M-258-002	Flow Diagram Turbine Plant Cooling Water System P&ID	7
11405-M-258-003	Flow Diagram Turbine Plant Cooling Water System P&ID	11

Section 1R05: Fire Protection

Procedures

Number	Title	Revision Date
EA-FC-89-055	Safe Shutdown Analysis	14

Section 1R11: Licensed Operator Requalification Program and Licensed Operator Performance

Procedures

Number	Title	Revision
SO-O-1	Conduct of Operations	101
OI-ST-10	Turbine Testing	60

Section 1R12: Maintenance Effectiveness

Procedures

Number	Title	Revision
PBD-16	Maintenance Rule Program Basis Document	9a
PED-SED-34	Maintenance Rule Program	9

Condition Reports (CRs)

2010-0296	2010-1174	2010-4278	2010-6150	2010-6774
2011-5149	2011-9945	2012-07204	2012-07225	2012-07469
2012-07473	2012-07474	2012-07475	2012-07476	2012-07477
2012-08058	2012-12140	2012-12143	2012-16590	2012-16597
2012-18878	2012-19783	2013-01093	2013-01107	2013-03435
2013-03671	2013-13008	2013-13269	2013-13291	2013-13354
2013-13971	2013-15135	2013-18603	2013-21158	2013-22468
2014-01943	2014-01969	2014-02036	2014-02960	

Miscellaneous Documents

Title	Revision / Date
Status of Equipment in MR Category (a)(1) or (a)(1) review	2/7/14
Functional Scoping Data Sheet for Auxiliary Feedwater Pumps	8a
Functional Scoping Data Sheet for Cooler Bypass Valves	2a
Functional Scoping Data Sheet for Stator Cooler	2a
Functional Scoping Data Sheet for Stator Cooling Water Sub-system	1a

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

Number	Title	Revision
FCSG-19	Performing Risk Assessments	17

Procedures

Number	Title	Revision
SO-M-100	Conduct of Maintenance	57b
SO-M-101	Maintenance Work Control	103

Section 1R15: Operability Determinations and Functionality Assessments

Procedures

Number	Title	Revision Date
PBD-9	Relief Valve Program	18
EC38435	Relief Valve Set Pressure Increase	0
WO270622		

Condition Reports (CRs)

2014-02666	2014-02667	2014-02791	2014-02900	2014-02941
2014-03054	2014-03055	2013-14592	2013-14212	2012-06370
2009-05608	2009-03256	200303990	2014-02838	2014-04096

Section 1R19: Post-Maintenance Testing

Procedures

Number	Title	Revision Date
QAP-6.2	Corrective Maintenance	5
SO-M-101	Maintenance Work Control	103
SO-O-1	Conduct of Operations	101
MD-AD-0004	Maintenance Work Instructions Writer’s Guide	38
MD-AD-0013	Post Maintenance Testing Selection Instructions	21
OP-ST-RW-3031	AC-10D Raw Water Pump Quarterly Inservice Test	40
OI-RW-1	Raw Water System Normal Operation	108
SO-M-100	Conduct of Maintenance	57b
IC-CP-01-5043	Calibration of Stator Cooling Water System Conductivity Elements and Recorder	5

Procedures

Number	Title	Revision Date
IC-ST-ESF-0005	Quarterly Functional Test of Pressurizer Pressure Low Signal P-102 Channels	7a
FC06850	Seismic & Weak Link Analsis for Masoneilan 20” Butrterfly Valves	0

Condition Reports (CRs)

2014-02255	2014-03084	2014-03091	2014-03142	2014-03192
2014-04175	2013-00195	2012-08134	2014-02360	2014-00472
2014-00522	2014-02363	200604627	200700756	2007-04901
2008-0146	2008-5585	2008-5586	2008-5587	2012-12084

Work Orders (WOs)

514425	471769

Section 4OA1: Performance Indicator Verification

Procedures

Number	Title	Revision
NOD-QP-37	Performance Indicator Program	27
NOD-QP-40	NRC Performance Indicator Program	8

Licensee Event Reports

Number	Title	Revision
2012-020	Raw Water Pump Anchors	0
2012-021	HCV-2987, HPSI Alternate Header Isolation Valve	0
2013-001	Mounting of GE HFA Relays Does Not Meet Seismic Requirements	0
2013-002	CVCS Class 1 & 2 Charging Supports are Unanalyzed	1
2013-003	Calculations Indicate the HPSI Pumps Operate in Run-out	1
2013-004	Inverters Inoperable During Emergency Diesel Generator Operation	0
2013-005	Control Room HVAC Modification Not Properly Evaluated	1
2013-006	Use of Teflon in LPSI and CS Pump Mechanical Seals	1

Licensee Event Reports

Number	Title	Revision
2013-007	Containment Air Cooling Units (VA-16A/B) Seismic Criteria	
2013-008	Previously Installed GE IAV Relays Failed Seismic Testing	1
2013-009	Tornado Missile Vulnerabilities	0
2013-010	HPSI Pump Flow Imbalance	1
2013-011	Inadequate Design for High Energy Line Break in Rooms 13 and 19 of the Auxiliary Building	0
2013-012	Intake Structure Crane Seismic Qualification	0
2013-014	Unqualified Components Used in Safety System Control Circuit	0
2013-017	Containment Spray Pump Design Documents do not Support Operation in Runout	0

Miscellaneous Documents

Number	Title	Revision
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	7

Condition Reports (CRs)

2012-01601	2013-01796	2013-02100	2013-03866	2013-04266
2013-04746	2013-15474	2013-05570	2013-18752	

Section 4OA2: Problem Identification and Resolution

Procedures

Number	Title	Revision
QAP-10.4	Condition Reporting and Corrective Action	14
	Corrective Action Trend Report (CAPTR)	1/31/2014
	Radiation Protection CAPTR	1/15/2014
	Chemistry CAPTR	1/8/2014
	Maintenance CAPTR	1/15/2014
	Training CAPTR	1/27/14
	Integrated Work Management CAPTR	1/15/14
	Security Services CAPTR	1/14/14

Procedures

Number	Title	Revision
	Operations CAPTR	1/10/14
	Nuclear Engineering Division CAPTR	1/9/14
FCSG-24-1	Condition Report Initiation	6
FCSG-24-3	Condition Report Screening	12a
FCSG-24-4	Condition Report and Cause Evaluation	8a
FCSG-24-5	Cause Evaluation Manual	7a
FCSG-24-6	Corrective Action Implementation and Condition Report Closure	12a
FCSG-24-10	Corrective Action Program Trending	5
SO-R-2	Condition Reporting and Corrective Action	53b

Condition Reports (CRs)

2014-02382	2014-03256	2014-03287	2014-03391	2014-03373
2014-03396	2014-03487	2014-03515	2013-17863	2013-23069
2013-22170	2013-16392	2013-22134	2013-23267	2013-15482
2013-19254	2013-19765	2014-01017	2013-21356	2013-21356
2013-22967	2014-00110	2013-16631	2014-03499	2014-02435
2014-01007	2014-03485	2011-10135	2013-08675	2014-01905
2014-00434	2014-00560	2014-01080	2014-01177	2014-01667
2014-02179	2014-02484	2014-02497	2014-02661	2014-03190
2014-03238	2014-01790	2014-01667	2014-01921	2014-01924
2014-01919	2014-01922	2014-01923	2014-01894	2014-01929
2014-02217	2014-02218	2014-01914	2014-02499	2014-01672
2014-02368	2014-02448	2014-01857	2014-02263	

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

Procedures

Number	Title	Revision
NOD-QP-3	10 CFR 50.59 and 10 CFR 72.48 Reviews	37
FCSG-23	10 CFR 50.59 Resource Manual	8
SO-R-1	Reportability Determinations	33

Condition Reports (CRs)

2014-00329	2014-00485

Section 4OA4: IMC 0350 Inspection Activities

CONDITION REPORTS (CR)

2012-4392

PROCEDURES

NUMBER	TITLE	REVISION
OI-SI-1	Operating Procedure, Safety Injection - Normal Operation	139

CALCULATIONS

NUMBER	TITLE	REVISION
FC8189	Evaluation of Operability, Containment Internal Structure	2
FC8235	Capacity calculation of beam B-22 using Finite Element Analysis	1
FC8314	ANSYS Analysis of Benchmark Beams	0
FC8159	Magnitude and Distribution of Loads Applied to Containment Internal Structure Outside the Steam Generator Compartment	2
FC6971	RV Head Laydown Area Seismic Analysis	2
FC8252	CIS Diaphragm Action	0
FC8164	Criteria and Methodology for Fort Calhoun Containment Internal Structure Analysis and Redesign	0
FC8159	Loads Applicable to the Fort Calhoun Containment Internal Structure Beams, Columns, and Slabs	1

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
ACI 318-63	American Concrete Standard 318: Building Code Requirements for Reinforced Concrete	1963
NUREG/CR-0098	Development of Criteria for Seismic Review of Selected Power Plants	1977

MISCELLANEOUS DOCUMENTS

ACI SPECIAL PUBLICATIONS 237	Vecchio, Frank J. "On the Post-Peak Ductility of Shear-Critical Beams."	(2006): 109