UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

May 10, 2011

David J. Bannister, 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 INTEGRATED INSPECTION REPORT NUMBER 05000285/2011002

Dear Mr. Bannister:

On March 31, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on April 7, 2011, with Mr. Jeffrey Reinhart, Site Vice President, and other members of your staff.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents three NRC-identified violations of very low safety significance (Green) and one NRC-identified Severity Level IV violation. All of these findings were determined to involve violations of NRC requirements. Additionally, one licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest the violations or the significance of the noncited violations, 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, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Fort Calhoun Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding 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 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosure, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary information so that it can be made available to the public without redaction.

Sincerely,

/RA/

Jeffrey A. Clark, P.E.
Chief, Project Branch E
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2011002
w/Attachment: Supplemental Information

cc w/Enclosure:

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2011002
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	January 1 through March 31, 2011
Inspectors:	J. Kirkland, Senior Resident Inspector J. Wingebach, Resident Inspector A. Fairbanks, Resident Inspector
Approved By:	Jeffrey Clark, P.E., Chief, Project Branch E Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2011002; 01/01/2011 – 03/31/2011; Fort Calhoun Station, Integrated Resident and Regional Report; Adverse Weather Protection, Refueling and Other Outage Activities, and Surveillance Testing.

The report covered a 3-month period of inspection by resident inspectors and one region-based inspector. Four Green noncited violations of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” The crosscutting aspect is determined using Inspection Manual Chapter 0310, “Components within the Cross Cutting Areas.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A self-revealing Green noncited violation of Fort Calhoun Station Technical Specification 5.8.1 occurred for an inadequate procedure for securing auxiliary feedwater flow when feeding the steam generators through the auxiliary feedwater ring. This inadequacy resulted in a complete loss of auxiliary feedwater for approximately three minutes. This was entered into the licensee’s corrective action program as Condition Report 2011-0839.

  The inspectors determined that the licensee’s inadequate operating instruction procedure was a performance deficiency. This finding was more than minor because it adversely impacted the human performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors performed the initial significance determination for the inoperable auxiliary feedwater system. The turbine-driven and motor-driven auxiliary feedwater pumps were inoperable for approximately three minutes, while the pump discharge lines were isolated during startup. The non-safety diesel-driven auxiliary feedwater pump remained available. The inspectors used the Inspection Manual 0609, Attachment 0609.04, “Phase 1 – Initial Screening and Characterization of Findings.” The finding screened to a Phase 2 significance determination because it involved an actual loss of safety function in the Mitigating Systems Cornerstone. A Region IV senior reactor analyst performed a Phase 2 significance determination and attempted to use the pre-solved worksheet from the “Risk Informed Inspection Notebook for Fort Calhoun Station,” Revision 2.01a. However, the pre-solved worksheet did not include the simultaneous failure of two auxiliary feedwater pumps. Therefore, the analyst performed a bounding Phase 3 significance determination. The analyst used the Fort Calhoun Standardized Plant Analysis Risk model, Revision 8.15, dated August 27, 2010, to calculate the conditional core damage probability, for a bounding event that included the failure to start for both the motor and turbine-driven auxiliary feedwater pumps. The change in core damage frequency was approximately 8.6×10^{-9}/year. This finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance (Section1R20).

- Green. The inspectors identified a Green noncited violation of 10 CFR 50 Appendix B Criterion XVI, "Corrective Actions," which states in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the this, between July 28, 2003, and November 29, 2010, the licensee failed to determine the cause of the out of tolerance condition impacting reactor protection system channel A trip unit 6, which was a significant condition adverse to qualty. This was entered into the licensee's corrective action program as Condition Report 2010-6190.

  The licensee's repeated failure to preclude the out-of-tolerance condition regarding reactor protection system channel A trip unit 6 is a performance deficiency. This finding is more than minor because if left uncorrected, the finding could have become more significant, in that, the licensee could fall below the technical specification "Minimum Operable Channels" if two additional trip unit six channels (B, C, or D) became inoperable. Because this finding occurred while the unit was operating at full power, the inspectors used Inspection Manual Chapter 0609 to determine its significance. Using Attachment 4 of that chapter, the inspectors determined that this finding has a very low safety significance (Green) because it was not a design or qualification deficiency, does not represent an actual loss of safety function, nor did it screen as potentially risk significant for external events. The finding was indicative of present performance and had a crosscutting aspect in the area of human performance associated with decision-making in that the licensee failed to use conservative assumptions in decision-making. The failure of the licensee to preclude repetition of the out-of-tolerance condition of reactor protection system channel A trip unit 6 is a significant condition adverse to quality. [H.1(b)] (Section 1R22)

- Green. The inspectors identified a Green noncited 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, since 1998, the licensee failed to verify the adequacy of the design of the safety injection refueling water tank vortex eliminator to prevent potential air entrainment due to vortexing in safety-related pump suction piping.

This finding was entered into the licensee's corrective action program as Condition Reports 2007-2452 and 2011-0311.

The inspectors determined that the failure to verify the adequacy of the safety injection refueling water tank vortex eliminator was a performance deficiency. The 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. Inspectors performed a Phase 1 screening, in accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee performed subsequent analysis which demonstrated that vortexing in the safety injection refueling water tank would not impact safety-related pump operation during a design basis event. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance (Section 4OA3).

Cornerstone: Miscellaneous

- Severity Level IV. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73 (a)(2)(i)(B) for the licensees' failure to submit a licensee event report within 60 days of discovery. On November 29, 2010, the licensee had the available information to determine reactor protection system channel A trip unit 6 had been inoperable from November 8 until November 29, 2010. Per the licensee's technical specifications, reactor protection system channel A trip unit 6 should have been in the tripped condition within 48 hours from time of discovering loss of operability. This is a reportable condition required by 10 CFR 50.73 (a)(2)(i)(B) as a condition prohibited by technical specifications. This was entered into the licensee's corrective action program as Condition Report 2011-2006.

  The inspectors determined that the licensees' failure to submit a licensee event report within the required time was a performance deficiency. The licensee had the appropriate licensing basis information as well as the inspector's specific concerns regarding inadequate troubleshooting, potential preconditioning, inadequate maintenance, and operability concern; therefore the performance deficiency was within their ability to foresee and correct. The inspectors reviewed this issue in accordance with Inspection Manual Chapter 0612 and the NRC Enforcement Manual. Through this review, the inspectors determined that traditional enforcement was applicable to this issue because the NRC's regulatory ability was potentially affected. Specifically, the NRC relies on the licensee to identify and report conditions or events meeting the criteria specified in regulations in order to perform its regulatory function, and when this is not done the regulatory function is impacted, and is therefore a finding. The inspectors determined that this finding was not suitable for evaluation using the significance determination process, and as such, was evaluated for traditional enforcement only, in accordance with the NRC Enforcement Policy. This is a Severity Level IV noncited violation consistent with Sections 2.3.2 and 6.9.d of the NRC Enforcement Policy. (Section1R22)

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

The unit began the assessment period at 100 percent power. On February 4, 2011, the unit shutdown to replace four trip contactors in the reactor protective system, and was restored to 100 percent power on February 6, 2011. On February 21, 2011, a power decrease was commenced due to two inoperable high power trip units. After restoring one trip unit, the power decrease was halted at 90 percent power, and the unit was restored to 100 percent power, on February 22, 2011, where it remained through the end of the assessment period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors performed a review of the adverse weather procedures for seasonal extremes (e.g., extreme high temperatures, extreme low temperatures, or hurricane season preparations). The inspectors verified that weather-related equipment deficiencies identified during the previous year were corrected prior to the onset of seasonal extremes, and evaluated the implementation of the adverse weather preparation procedures and compensatory measures for the affected conditions before the onset of, and during, the adverse weather conditions.

During the inspection, the inspectors focused on plant-specific design features and the procedures used by plant personnel to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Safety Analysis Report and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. Specific documents reviewed during this inspection are listed in the attachment. The inspectors also reviewed corrective action program items to verify that plant personnel were identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the following plant systems:

- Raw Water, Circulating Water, and Component Cooling Water Systems

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

These activities constitute completion of one (1) readiness for seasonal adverse weather sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- January 14, 2011, Portions of the 125 VDC System
- March 19, 2011, Portions of the Diesel Fire Pump System while maintenance was being performed on breaker 1A31
- March 30, 2011, Portions of the Auxiliary Feedwater System due to scaffolding concerns in the area of the electric feedwater pump, FW-6

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Safety Analysis Report, technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also inspected accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three (3) partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

On February 10, 2011, the inspectors performed a complete system alignment inspection of the containment spray system to verify the functional capability of the system. The inspectors selected this system because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors inspected the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. The inspectors reviewed a sample of past and outstanding work orders to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program database to ensure that system equipment-alignment problems were being identified and appropriately resolved. Specific documents reviewed during this inspection are listed in the attachment.

This inspection effort counts towards the completion of Temporary Instruction TI-2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems."

These activities constitute completion of one (1) complete system walkdown sample as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors conducted fire protection walkdowns that were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- January 31, 2011, Fire Area 3, Spent Regenerant Tank & Pump Area (Room 23)
- January 31, 2011, Fire Area 13, Mechanical Penetration Area (Room 13)
- January 31, 2011, Fire Area 20.7, Transfer Canal Pump Room (Room 24)
- March 14, 2011, Fire Area 41, Cable Spreading Room (Room 70)

The inspectors reviewed areas to assess if licensee personnel had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features, in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to affect equipment that could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's corrective action program. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four (4) quarterly fire-protection inspection samples as defined in Inspection Procedure 71111.05-05.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the raw water/component cooling water heat exchanger, AC-1B. The inspectors verified that performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; the licensee utilized the periodic maintenance method outlined in EPRI Report NP 7552, "Heat Exchanger Performance Monitoring Guidelines"; the licensee properly utilized biofouling controls; the licensee's heat exchanger inspections adequately assessed the state of cleanliness of their tubes; and the heat exchanger was correctly categorized under 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) heat sink inspection sample as defined in Inspection Procedure 71111.07-05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On March 8, 2011, the inspectors observed a crew of licensed operators in the plant's simulator to verify that operator performance was adequate, evaluators were identifying, and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance
- Crew's clarity and formality of communications
- Crew's ability to take timely actions in the conservative direction
- Crew's prioritization, interpretation, and verification of annunciator alarms
- Crew's correct use and implementation of abnormal and emergency procedures
- Control board manipulations
- Supervisors oversight and direction
- Crew's ability to identify and implement appropriate technical specification actions and emergency plan actions and notifications

The inspectors compared the crew's performance in these areas to pre-established operator action expectations and successful critical task completion requirements. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) quarterly licensed-operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk significant systems:

- Control Room Air Conditioner, VA-46A
- Clutch power supply system associated with the Reactor Protective System

The inspectors reviewed events such as where ineffective equipment maintenance has resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or (a)(2)
- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two (2) quarterly maintenance effectiveness sample as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed licensee personnel's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- January 18, 2011, Risk management actions associated with auxiliary feedwater pump FW-10 while diesel generator 1 was out of service

- March 14, 2011, Orange activity risk with diesel generator 1 out of service for a mini-overhaul

- March 21, 2011, Risk management actions associated with the failure of breaker 1A31

The inspectors selected these activities based on potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three (3) maintenance risk assessments and emergent work control inspection samples as defined in Inspection Procedure 71111.13-05.

b. <u>Findings</u>

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. <u>Inspection Scope</u>

The inspectors reviewed the following issues:

- February 14, 2011, Operability of auxiliary feedwater pump FW-6 while auxiliary feedwater pump FW-10 had a degraded discharge check valve

- February 20, 2011, Operability of raw water pump AC-10B following discovery of a degraded seismic restraint

- February 21, 2011, Operability of the reactor protective system following simultaneous failures of A and D channel high power trip units

- March 25, 2011, Operability of both diesel generators following discovery of cracked welds on the oil air bath supports

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and Updated Safety Analysis Report to the licensee personnel's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four (4) operability evaluations inspection samples as defined in Inspection Procedure 71111.15-04.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

To verify that the safety functions of important safety systems were not degraded, the inspectors reviewed the temporary modification identified as removal of high winding temperature sensors from control room air conditioner VA-46A.

The inspectors reviewed the temporary modification and the associated safety-evaluation screening against the system design bases documentation, including the Updated Safety Analysis Report and the technical specifications, and verified that the modification did not adversely affect the system operability/availability. The inspectors also verified that the installation and restoration were consistent with the modification documents and that configuration control was adequate. Additionally, the inspectors verified that the temporary modification was identified on control room drawings, appropriate tags were placed on the affected equipment, and licensee personnel evaluated the combined effects on mitigating systems and the integrity of radiological barriers.

These activities constitute completion of one (1) sample for temporary plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following postmaintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- January 14, 2011, Component cooling water pump AC-3C after pump rebuild

- January 19, 2011, Control room air conditioner VA-46A after switch replacement and troubleshooting

- February 5, 2011, Functional test of reactor protective system trip logic following replacement of the M-Contactors

- February 5, 2011, Reactor manual trip test following replacement of the M-Contactors

- February 22, 2011, Postmaintenance testing following replacement of A-Channel reactor protective system power supply

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. The inspectors evaluated these activities for the following (as applicable):

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed

- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specifications, the Updated Safety Analysis Report, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of five (5) postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. <u>Findings</u>

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. <u>Inspection Scope</u>

The inspectors reviewed the outage safety plan and contingency plans for the maintenance outage conducted February 4-6, 2011, to confirm that licensee personnel had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense in depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

- Configuration management, including maintenance of defense in depth, is commensurate with the outage safety plan for key safety functions and compliance with the applicable technical specifications when taking equipment out of service.

- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.

- Status and configuration of electrical systems to ensure that technical specifications and outage safety-plan requirements were met, and controls over switchyard activities.

- Verification that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.

- Controls over activities that could affect reactivity.

- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left, which could block emergency core cooling system suction strainers, and reactor physics testing.

- Licensee identification and resolution of problems related to outage activities.

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

These activities constitute completion of one (1) refueling outage and other outage inspection sample as defined in Inspection Procedure 71111.20-05.

b. Findings

Introduction. A self-revealing Green noncited violation of Fort Calhoun Station Technical Specification 5.8.1 occurred for an inadequate procedure for securing auxiliary feedwater flow when feeding the steam generators through the auxiliary feedwater ring. This inadequacy resulted in a complete loss of auxiliary feedwater for approximately three minutes.

Description. On February 5, 2011, a reactor startup was in progress. During startup, the licensee would normally feed the steam generators through the main feedwater ring, using the non-safety related diesel-driven auxiliary feedwater pump, FW-54. However, prior to reactor startup, the hydrazine injection pump, CF-13, failed. This pump is used for secondary chemistry control while feeding the steam generators using FW-54.

With no hydrazine injection available for FW-54, the steam generators were fed using the electric auxiliary feedwater pump, FW-6. Hydrazine is added to the emergency feedwater storage tank, which allows secondary chemistry control while using pump FW-6. To feed the steam generators with pump FW-6, the operators followed Operating Instruction OI-AFW-4, "Auxiliary Feedwater Startup and System Operation." The operators had not specifically trained on this operating instruction during just in time training prior to the reactor startup, and no detailed pre-job brief was conducted prior to implementing the procedure. The general flow-path is from the emergency feedwater storage tank, through pump FW-6, through the auxiliary feedwater inlet valves HCV-1107A/B and HCV-1108A/B into steam generators A and B respectively. During this operation, HCV-1107B and HCV-1108B hand controllers were in the "auto" position, and HCV-1107A and HCV-1108A were in their "open" position. Flow to the steam generators was being controlled by the Hand Instrument Controllers for HCV-1107B and HCV-1108B.

After the main feed pumps were started and steam generators were being fed by the main feed, the auxiliary feed operation was secured using Procedure OI-AFW-4. Attachment 3, Step 6.d of the procedure states "Ensure both of the following are closed: HCV-1107A and HCV-1108A," which the operator had completed. However, with the valves in the closed position, they could not automatically open on a valid auxiliary feed water actuation signal, therefore rendering both trains of auxiliary feedwater inoperable. Once the operators recognized the condition (approximately three minutes), they placed the valve switches for HCV-1107A and HCV-1108A into the AUTO position, restoring operability of the auxiliary feedwater system. The licensee entered this issue into their corrective action program as Condition Report 2011-0839.

Analysis. The inspectors determined that the licensee's inadequate operating instruction procedure was a performance deficiency. This finding was greater than minor because it adversely impacted the human performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors performed the initial significance determination for the inoperable auxiliary feedwater system. The turbine-driven and motor-driven auxiliary feedwater pumps were inoperable for approximately three minutes, while the pump discharge lines were isolated during startup. The non-safety diesel-driven auxiliary feedwater pump remained available. The inspectors used the Inspection Manual 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The finding screened to a Phase 2 significance determination because it involved an actual loss of safety function in the Mitigating Systems Cornerstone. A Region IV senior reactor analyst performed a Phase 2 significance determination and attempted to use the pre-solved worksheet from the "Risk Informed Inspection Notebook for Fort Calhoun Station," Revision 2.01a. However, the pre-solved worksheet did not include the simultaneous failure of two auxiliary feedwater pumps. Therefore, the analyst performed a bounding Phase 3 significance determination. The analyst used the Fort Calhoun Standardized Plant Analysis Risk model, Revision 8.15, dated August 27, 2010, to calculate the conditional core damage probability, for a bounding event that included the failure to start for both the motor and turbine-driven auxiliary feedwater pumps. While the pumps were likely recoverable, the analyst conservatively assumed that they were not recoverable. The analyst used a truncation limit of $1x10^{-11}$. The conditional core damage probability for a one year exposure period was $1.5x10^{-3}$. The analyst noted that the nominal case core damage frequency was $1.0x10^{-5}$/year. The incremental conditional core damage probability for the one year exposure was therefore $1.5x10^{-3}$. For the three minute exposure period, the bounding delta-core damage frequency (CDF) was:

Delta-CDF = $1.5x10^{-3}$/year * 3 minutes/[(8760hours/year)*(60 minutes/hour)] = $8.6x10^{-9}$/year

Since the delta-CDF was very low, the analyst qualitatively determined that external events were not a significant contributor to delta-CDF. In addition, since the delta-CDF was less that $1x10^{-7}$/year, the analyst determined that the finding was not a significant contributor to the large early release frequency. The finding was of very low safety significance (Green).

The dominant core damage sequences included a spurious steam generator isolation signal initiating event; the failure of the diesel-driven, motor-driven and turbine-driven auxiliary feedwater pumps; and the failure of operators to initiate feed and bleed. The very short exposure period coupled with the functional diesel-driven auxiliary feedwater pump helped to mitigate the significance. This finding did not have a cross-cutting aspect because the most significant contribution did not reflect current licensee performance.

<u>Enforcement</u>. Fort Calhoun Station Technical Specification 5.8.1, requires, in part, that the licensee establish and implement written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978, which includes procedures for startup of the auxiliary feedwater system. Contrary to the above, Operating Instruction OI-AFW-4, "Auxiliary Feedwater Startup and System Operation," did not adequately ensure that the plant could respond to an auxiliary feedwater actuation signal during the shutdown sequence of the procedure. This inadequacy resulted in a complete loss of auxiliary feedwater on February 5, 2011. Because the violation was of very low safety significance (Green) and was entered into the licensee's corrective action

program as Condition Report 2011-0839, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2011-002-01, “Inadequate Operating Instruction Results in a Loss of Auxiliary Feedwater.”

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

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

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

- January 13, 2011, Personnel Access Lock O-Ring Seal Test
- January 21, 2011, Operability Test of IA-YCV-1045-C and Close Stroke Test of YCV-1045
- January 27, 2011, Reactor Coolant System Leak Rate Test
- February 10, 2011, Quarterly Functional Test of Steam Generator Low Pressure and Asymmetric Steam Generator Transient Reactor Protection System Bi-stable Trip Units
- February 16, 2011, Auxiliary Feedwater Pump FW-10, Steam Isolation Valve, and Check Valve Tests
- March 18, 2011, Diesel Generator 1 Check

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

These activities constitute completion of six (6) surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

(1) Failure to Determine the Cause of the Out-of-Tolerance Condition Regarding Reactor Protection System Channel A Trip Unit 6

Introduction. The inspectors identified a Green noncited violation of 10 CFR 50 Appendix B Criterion XVI , "Corrective Actions," states in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the this, between July 28, 2003, and November 29, 2010, the licensee failed to determine the cause of the out of tolerance condition impacting reactor protection system channel A trip unit 6 (A/TU-6).

Description. On November 8, 2010, the licensee performed Surveillance Test IC-ST-RPS-0044, calibration of steam generator low-pressure trip unit A/TU-6 and asymmetrical steam generator transient trip unit A/TU-7. During the performance of this test, out-of-tolerance as-found values were recorded at approximately 10:30 a.m. Specifically, on Attachment 9.1 - Data Sheet 1, terminal 73 and terminal 75 as found values were elevated out of tolerance by 0.1 millivolt dc and 0.2 millivolt dc respectively. These values normally correspond to RC-2A and RC-2B steam generator pressure. RC-2A and RC-2B pressure transmitters scale from 1-5 volts corresponding to 0 - 1000 psi. The condition observed was non-conservative, in that, increased voltage would mask a low-pressure condition to the reactor protection system by a linear amount. The intent of gathering this data is to ensure that there is minimal degradation of the signals by the circuitry prior to trip unit input. The remaining as-found data required by the surveillance test was recorded by the instrumentation and controls technician. The remaining values including trip unit A/TU-6 and trip unit A/TU-7 input values were in specification. The out-of-tolerance values failed the surveillance test acceptance criteria. Work Request 157517 was generated to troubleshoot and repair the out-of-tolerance condition. The licensees' Condition Report 2010-5645 documents the failed surveillance test.

Terminal 74 is connected to common and should have had a value of zero Vdc. Instead, this terminal was reading greater than zero Vdc. Common to all three terminals is AI-31A-AW12 B2 contact module, which is part of the asymmetrical steam generator transient test circuit and should not affect the trip unit circuits. During trouble shooting efforts, it was determined the issue resided within the asymmetrical steam generator transient test circuit. Instrumentation and controls technicians knew this module had previously been an issue. Condition Reports 200302822 and 2009-2317 document past out-of-tolerance results. Cycling the contact module or replacing it had cleared the out-of-tolerance values in the past; therefore part of FC-1212 troubleshooting plan was to cycle the contact module. The FC-1212 was executed and no maintenance activities were performed.

The licensee reperformed surveillance Test IC-ST-RPS-0044 to check the required values. The out-of-tolerance values were now in-tolerance. The on-shift instrumentation and controls technician did not intend to complete the surveillance test. Instead, the trip units were left in bypass and the results were discussed with the shift manager including a safety concern regarding the contact module. This concern was documented in Condition Report 2010-5667 on November 8, 2010, at 3:00 p.m. The condition report questioned the problem with the contact module and stated that if the problem occurred again there would be no indication to the control room. It also stated that the asymmetrical steam generator transient test relay was exercised during troubleshooting specifically to make a better connection to pass the IC-ST-RPS-0044 surveillance test. At approximately 5:06 p.m., the night shift instrumentation and controls technician completed the last three steps of IC-ST-RPS-0044 with the day shift operations crew based on the data recorded by the day shift instrumentation and controls technician. This consisted of ensuring the trip units were reset, removing the bypass keys, and informing the shift manager. The trip units were returned to service and an operability determination was requested by the shift manager to evaluate the asymmetrical steam generator transient test circuit during normal operation for operability.

Based on discussions with instrumentation and controls personnel and the shift manager, as well as reviewing condition reports, the inspectors' questioned if (1) the surveillance test used to declare operability had been compromised due to potential preconditioning, and (2) what corrective actions were taken to correct the problem. Subsequently, on November 10, 2010, the licensee documented these questions into Condition Report 2010-5733. Based on discussions with licensing and instrumentation and controls personnel the operating crew declared trip units A/TU-6 and A/TU-7 inoperable, replaced the contact module, performed Surveillance Test IC-ST-RPS-0044 again, and then returned the trip units to service.

On November 16, 2010, operability determination associated with Condition Report 2010-5667 was completed. It concluded that the out-of-tolerance values regarding A T/U-6 from November 8, 2010, were not outside the design basis, as the values do not account for a 4 psi margin not built into the tolerances based on Calculation FC05733. Therefore, the values could be out-of-tolerance +/- 16 millivolt dc before they are outside of their design basis. In addition, the increase in voltage does not affect trip unit A/TU-7 as the voltage is added to each signal, which are then subtracted to determine a difference. To address the concern regarding the asymmetrical steam generator transient test circuit effect on trip unit operability additional actions were required to confirm operability in the current calibration cycle. Specifically, Work Order 396853 was generated to monitor the voltage of the relay contact on all channels to confirm operability. Surveillance test IC-ST-RPS-0044 test frequency was increased for the next six weeks.

On November 29, 2010, voltage at terminal 74 was elevated 39-millivolt dc, thus rendering trip unit A/TU-6 inoperable. This is documented in the operator logs as well as in Condition Report 2010-6190. Trip Unit A/TU-6 was declared inoperable. Subsequent trouble shooting determined a bad wire in the circuit. The wire was replaced, post-maintenance testing performed, and the trip unit returned to service.

Condition Reports 200302822 and 2009-2317 documents prior out-of-tolerance readings, for the same values in Surveillance test IC-ST-RPS-0044, which rendered the trip unit inoperable. These events were not determined by the licensee to be functional failures. After reviewing the condition reports, the inspectors believe these particular events to be functional failures of trip unit A/TU-6.

The events on November 29, 2010, show that the corrective actions taken in response to the out-of-tolerance conditions on July 28, 2003; May 14, 2009 and November 8, 2010, were inadequate. These actions consisted of replacing the AI-31A-AW12 B2 contact module. The condition was repetitive, and therefore, within the licensee's ability to foresee and correct.

The actions taken in response to the events on November 8 and November 10, 2010, were inadequate as demonstrated when trip unit A/TU-6 was declared inoperable on November 29, 2010, which the licensee documented in Condition Report 2010-6190.

Analysis. The licensee's repeated failure to determine the cause of the out-of-tolerance condition regarding reactor protection system channel A trip unit 6 is a performance deficiency. The finding is more than minor because if left uncorrected, the finding could have become more significant, in that, the licensee could fall below the technical specification "Minimum Operable Channels" if two additional trip unit six channels (B, C, or D) became inoperable. Because this finding occurred while the unit was operating at full power, the inspectors used Inspection Manual Chapter 0609 to determine its significance. Using Attachment 4 of that chapter, the inspectors determined that this finding has a very low safety significance (Green) because it was not a design or qualification deficiency, does not represent an actual loss of safety function, nor did it screen as potentially risk significant for external events. The finding was indicative of present performance and had a crosscutting aspect in the area of human performance associated with decision-making in that the licensee failed to use conservative assumptions in decision making [H.1(b)] The failure of the licensee to preclude repetition of the out-of-tolerance condition of reactor protection system channel A trip unit 6 is a significant condition adverse to quality. Title 10 CFR 50 Appendix B, Criterion XVI requires that measures be established to promptly identify and correct conditions adverse to quality. Repeatedly replacing the contact module is not a fix unless the contact module was deficient, which it was not.

Enforcement. Criterion XVI of Appendix B to Section 50 of Title 10 of the Code of Federal Regulations states, in part, that conditions adverse to quality "…are promptly identified and corrected." Contrary to the above, since July 28, 2003, the licensee repeatedly failed to correct the inoperable condition of reactor protection system channel A trip unit 6 due to a defective wire. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as Condition Report 2010-6190, this violation is being treated as a noncited violation, consistent with the 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2011-002-02, "Failure to Determine the Cause of the Out Of Tolerance Condition Regarding Reactor Protection System Channel A Trip Unit 6."

(2) Failure to Submit a Timely Licensee Event Report

Introduction. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73 (a)(2)(i)(B) for the licensees' failure to realize an event was reportable and therefore submit a licensee event report within 60 days of discovery. On November 29, 2010, the licensee had the available information to determine reactor protection system A T/U-6, had been inoperable from November 8 until November 29, 2010. Per the licensee's technical specifications, reactor protection system A T/U-6 should have been in the trip position within 48 hours from time of discovering loss of operability. This is a reportable condition required by 10 CFR 50.73 (a)(2)(i)(B) as a condition prohibited by technical specifications.

Description. On November 8, 2010, the licensee performed Surveillance Test IC-ST-RPS-0044, calibration of steam generator low-pressure trip unit A/TU-6 and asymmetrical steam generator transient trip unit A/TU-7. During the performance of the surveillance test, as-found out-of-tolerance values were recorded at approximately 10:30 a.m. and documented in Condition Report 2010-5645.

During troubleshooting efforts it was incorrectly determined that the issue resided with the test circuit contactor box. Condition Reports 200302822 and 2009-2317 document past out of tolerance results. Cycling the contact module or replacing it had cleared out of tolerances values in the past, therefore part of the FC-1212 troubleshooting plan was to cycle the contact module. The FC-1212 was executed and no maintenance activities were performed. Cycling the asymmetrical steam generator transient test circuit was inappropriate to correct this condition adverse to quality per 10 CFR 50 Appendix B, Criterion XVI as the licensee had prior history that showed this would not in fact ultimately correct the condition.

Surveillance Test IC-ST-RPS-0044 was performed again to check the required values for change. The out of tolerance values were now in tolerance. The on-shift instrumentation and control technician did not intend to complete the surveillance test. Instead, the trip units were left in bypass and the results were discussed with the shift manager including a safety concern regarding the contact module. This concern was documented in Condition Report 2010-5667 on November 8, 2010, at 3:00 p.m. The condition report questioned the problem with the contact module and stated that if the problem occurred again there would be no indication to the control room. It also stated that the asymmetrical steam generator transient test relay was exercised during troubleshooting specifically to make a better connection to pass the IC-ST-RPS-0044 surveillance test. At approximately 5:06 p.m., the night shift instrumentation and controls technician completed the last three steps of Surveillance Test IC-ST-RPS-0044 with the day shift operations crew based on the data recorded by the day shift instrumentation and controls technician. This consisted of ensuring the trip units were reset, removing the bypass keys, and informing the shift manager. The trip units were returned to service and an operability determination was requested by the shift manager to evaluate the asymmetrical steam generator transient test circuit during normal operation for operability.

Based on discussions with instrumentation and controls personnel and the shift manager, as well as reviewing condition reports, the inspectors' questioned if (1) the surveillance test used to declare operability had been compromised due to potential preconditioning, and (2) what corrective actions were taken to correct the problem. Subsequently, on November 10, 2010, the licensee documented these questions into Condition Report 2010-5733. Based on discussions with licensing and instrumentation and controls personnel the operating crew declared trip units A/TU-6 and A/TU-7 inoperable, replaced the contact module, performed Surveillance Test IC-ST-RPS-0044 again, and then returned the trip units to service.

Based on discussions with licensing and instrumentation and controls personnel the operating crew declared trip units A/TU-6 and A/TU-7 inoperable, replaced the contact module, reperformed Surveillance Test IC-ST-RPS-0044, and then returned the trip units to service. Replacing the contact module was inappropriate to correct this condition adverse to quality per 10 CFR 50 Appendix B, Criterion XVI as the licensee had prior history that showed this would not in fact ultimately correct the condition. In addition the past contact modules that were replaced had not been analyzed to determine if they were deficient.

On November 16, 2010, the operability determination associated with Condition Report 2010-5667 was completed. This determination concluded that the out-of-tolerance values on November 8, 2010, were not outside the design basis as the values do not account for a 4 psi margin not built into the tolerances based on Calculation FC05733. Therefore, the values could be out of tolerance +/- 16-millivolt dc before they are outside of their design basis. In addition, the increase in voltage does not affect trip unit A/TU-7 as the voltage is added to each signal, which are then subtracted to determine a difference. To address the concern regarding the asymmetrical steam generator transient test circuit effect on trip unit operability additional actions were required to confirm operability in the current calibration cycle. Specifically Work Order 396853 was generated to monitor the voltage of the relay contact on all channels to confirm operability. Surveillance Test IC-ST-RPS-0044 test frequency was increased for the next six weeks.

On November 29, 2010, the licensee found that the voltage at terminal 74 was elevated 39-millivolt dc, thus rendering trip unit A/TU-6 inoperable. This condition was documented in the operator logs and in Condition Report 2010-6190. The licensee declared A T/U-6 inoperable. Subsequent trouble shooting determined a bad wire in the circuit. The wire was replaced, postmaintenance testing was performed, and the trip unit was returned to service. These events prove that the condition-effecting A T/U-6 was not corrected on November 8, 2010 and therefore should not have been returned to service.

Analysis. The inspectors determined that the licensees' failure to realize that an event was reportable and therefore submit a licensee event report within the required time was a performance deficiency. The licensee had the appropriate licensing basis information as well as the inspector's specific concerns regarding inadequate troubleshooting, potential preconditioning, inadequate maintenance, and operability concern; therefore the performance deficiency was within their ability to foresee and correct. The inspectors reviewed this issue in accordance with Inspection Manual Chapter 0612 and the NRC Enforcement Manual. Through this review, the inspectors determined that traditional enforcement was applicable to this issue because the NRC's regulatory ability was potentially affected. Specifically, the NRC relies on the licensee to identify and report conditions or events meeting the criteria specified in regulations in order to perform its regulatory function, and when this is not done the regulatory function is impacted, and is therefore a finding. The inspectors determined that this finding was not suitable for evaluation using the significance determination process, and as such, was evaluated for traditional enforcement only, in accordance with the NRC Enforcement Policy. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section 2.3.2. and 6.9.d of the Enforcement Policy

Enforcement. Title 10 CFR 50.73 (a)(2)(i)(B) requires in part, that the licensee shall submit a licensee event report within 60 days of any "…operation or condition which was prohibited by the plant's technical specifications". Contrary to the above, the licensee failed to submit a licensee event report within 60 days after they determined the reactor protection system channel A trip unit 6 was inoperable and should have realized it had been inoperable since November 8, 2011. There was no actual or potential safety consequences associated with this violation. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section 2.3.2 and 6.9.d of the Enforcement Policy. Since this violation is in the licensee's corrective action program as Condition Report 2011-2006, this violation is being treated as a noncited violation, consistent with NRC Enforcement Policy: NCV 05000285/2011002-03, "Failure to Submit a Timely Licensee Event Report."

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on February 8, 2011, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the Technical Support Center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness, with those identified by the licensee staff, in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the attachment.

These activities constitute completion of one (1) sample as defined in Inspection Procedure 71114.06-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Reactor Coolant System Specific Activity (BI01)

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system specific activity performance indicator for the period from the first quarter 2010 through the fourth quarter 2010. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's reactor coolant system chemistry samples, technical specification requirements, issue reports, event reports, and NRC integrated inspection reports for the period of January through December, 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) reactor coolant system specific activity sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.2 Reactor Coolant System Leakage (BI02)

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system leakage performance indicator for the period from the first quarter 2010 through the fourth quarter 2010. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator logs; reactor coolant system leakage tracking data, issue reports, event reports, and NRC integrated inspection reports for the period of January through December, 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one (1) reactor coolant system leakage sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

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

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Selected Issue Follow-up Inspection

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors recognized a corrective action item documenting a declining trend in the leakage past auxiliary feedwater pumps discharge check valves. The inspectors also reviewed an issue relating to increased gasket bleedoff on the reactor coolant pumps.

These activities constitute completion of two (2) in-depth problem identification and resolution samples as defined in Inspection Procedure 71152-05.

b. Findings

No findings were identified.

.4 In-depth Review of Operator Workarounds

a. Inspection Scope

The inspectors selected this issue for review to verify that licensee personnel were identifying operator workaround problems at an appropriate threshold and entering them in the corrective action program, and has proposed or implemented appropriate corrective actions. The inspectors considered the following, as applicable, during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

These activities constitute completion of one (1) in-depth review of operator workaround sample as defined in Inspection Procedure 71152-05.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 (Closed) Unresolved Item 05000285/2007007-07, Safety Injection Refueling Water Tank Vortexing

a. Inspection Scope

During a triennial component design basis inspection, documented in Inspection Report 05000285/2007007, the team identified that the licensee did not have analytical evidence or test results to verify the design of the safety injection refueling water tank vortex eliminator. The purpose of the vortex eliminator is to prevent air entrainment in the safety injection pumps when the water level in the tank is at its minimum. The failure to verify the adequacy of the safety injection refueling water tank vortex eliminator to prevent potential air entrainment due to vortexing in safety-related pump suction piping was a violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control. The licensee's subsequent analysis concluded that air entrainment would not adversely impact safety-related pump operation during a design basis event. This Unresolved Item is closed.

b. Findings

Failure to Verify Design Adequacy of Refueling Water Tank Vortex Eliminator

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, after the licensee failed to verify the adequacy of the safety injection refueling water tank vortex eliminator to prevent potential air entrainment due to vortexing in safety-related pump suction piping.

Description. During a triennial component design basis inspection, documented in Inspection Report 05000285/2007007, the team identified that the licensee did not have analytical evidence or test results to verify the design of the safety injection refueling water tank vortex eliminator. The purpose of the vortex eliminator is to prevent air entrainment in the safety injection pumps when the water level in the tank is at its minimum. In response to this issue, the licensee evaluated the adequacy of the safety injection refueling water tank vortex eliminator and documented the results in Condition Reports 2007-2452 and 2011-0311.

During the review of the licensee's analysis to address the unresolved item, the inspectors identified that the licensee had failed to assume single active failure of one low pressure safety injection pump to trip off after the initiation of a recirculation actuation signal. The failure to account for this single active failure impacted the licensee's conclusion that the refueling water tank minimum level was adequate to prevent air entrainment in safety-related pump suction piping. If a low pressure safety injection pump failed to trip off with the initiation of a recirculation action signal, the refueling water tank would be drained further than assumed in the accident analysis. The licensee originally told inspectors that the failure of the low pressure safety injection pump to trip off would not drain the tank further because the refueling water tank suction valve closes upon the initiation of a recirculation actuation signal. Inspectors informed the licensee that because the maximum stroke time for the valve was 25 seconds, the analysis would not adequately reflect all tank losses if the analysis assumed instantaneous valve realignment at the initiation of a recirculation actuation signal. The licensee re-performed the calculation with the appropriate single failure assumptions and concluded that the level of the safety injection refueling water tank would not drop below analyzed limits.

The licensee's evaluation included performing hydraulic scaled-model tests for the safety injection refueling water tank to study the potential for significant air intrusion. The results of the scaled-model tests indicated that safety-related pump operation would not be impacted by potential air entrainment due to vortexing.

Inspectors evaluated the methodology, assumptions, and calculations associated with verifying that air entrainment due to vortexing in the safety injection refueling water tank would not impact safety-related pump function. Specifically, inspectors noted that the licensee appropriately evaluated inventory loss from the refueling water tank during suction valve realignment. Additionally, inspectors reviewed the scaled-model test data and confirmed that sufficient evidence existed to support the licensee's conclusions that safety-related pump performance would not be impacted by significant air entrainment during a design basis event. Inspectors concluded that the licensee appropriately verified the adequacy of the safety injection refueling water tank vortex eliminator.

Analysis. The inspectors determined that the failure to verify the adequacy of the safety injection refueling water tank vortex eliminator was a performance deficiency. The 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. Inspectors performed a Phase 1 screening, in accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee performed subsequent analysis which demonstrated that vortexing in the safety injection refueling water tank would not impact safety-related pump operation during a design basis event. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The inspectors identified a Green noncited 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 above, the licensee failed to assure that design control measures were provided 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, since 1998, the licensee failed to verify the adequacy of the design of the safety injection refueling water tank vortex eliminator to prevent potential air entrainment due to vortexing in safety-related pump suction piping. This finding was entered into the licensee’s corrective action program as Condition Reports 2007-2452 and 2011-0311. Because this finding was determined to be of very low safety significance and was entered into the licensee’s corrective action program, this violation is being treated as a noncited violation consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2011002-04, “Failure to Verify Design Adequacy of Refueling Water Tank Vortex Eliminator.”

.2 (Open and Closed) Licensee Event Report 05000285/2010-005-01: Inoperability of the Emergency Diesel Generator Fuel Oil Transfer System

Diesel Fuel Oil Transfer Pump FO-37 and its credited portable back-up pump were inoperable on January 6-7, 2010. On January 6, 2010, FO-37 was rendered inoperable due to local area flooding caused by the rupture of FP-772, "Service Building Fire Sprinkler Isolation Valve.” The function of FO-37 is to transfer diesel fuel between "Diesel Fuel Oil Storage Tanks" FO-10 and FO-1. On June 24, 2010, an engineering evaluation determined that the credited portable back up pump to FO-37 was not the correct pump for the application and would not transfer diesel fuel oil from FO-10 to FO 1 as intended. Since both pumps (FO-37 and the credited portable back-up pump) were inoperable, this is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B). This licensee event report is a revision to the initial report to discuss the licensee’s root cause analysis.

A root cause analysis determined there was a failure to perform an appropriate design change evaluation for maintaining diesel fuel oil transfer system capability as required by Technical Specification Amendment 162, dated March 29, 1994. The originally credited portable back-up pump was replaced with a new portable pump that has the capability to transfer diesel fuel oil between tanks F0-10 and F0-1. The appropriate procedures were revised to address the pump replacement.

The revised licensee event report was reviewed by the inspectors, no findings of significance were identified, and no violation of NRC requirements occurred. This licensee event report is closed.

.3 (Open) Licensee Event Report 05000285/2010-006-00: Reactor Trip Due to Erroneous Moisture Separator Trip Signal

Fort Calhoun Station was operating at full power (nominal 100 percent). The station was preparing a scaffolding for an upcoming outage when on December 23, 2010, at 1050 Central Standard Time (CST), a reactor trip occurred. The operators entered Emergency Operating Procedure (EOP) 00, "Standard Post Trip Actions." The main steam and feedwater system operated normally. All control rods inserted fully. The apparent cause of the turbine and subsequent reactor trip was the inadvertent actuation, caused by bumping, and sticking of one of four turbine moisture separator high water level turbine trip switches while reactor power was above 15 percent. The root cause investigation is in progress. Following the initial determination of the erroneous moisture separator high level trip signal, immediate actions included: halting all work near the moisture separator sensing lines and level switches, posting the affected areas as "Protected Equipment," and initiating a stop work action for all ongoing scaffold work within the turbine building.

.4 <u>(Open) Licensee Event Report 05000285/2011-001-00: Inadequate Flooding Protection Due To Ineffective Oversight</u>

During identification and evaluation of flood barriers, unsealed through wall conduit penetrations in the outside wall of the intake structure were identified that are below the licensing basis flood elevation. A summary of the root causes included: a weak procedure revision process; insufficient oversight of work activities associated with external flood matters; ineffective identification, evaluation and resolution of performance deficiencies related to external flooding; and "safe as is" mindsets relative to external flooding events. The penetrations were temporarily sealed and a configuration change was developed and implemented whereby permanent seals were installed. Comprehensive corrective actions to address the root and contributing causes are being addressed through the corrective action program.

4OA6 Meetings

<u>Exit Meeting Summary</u>

On February 10, 2011, the inspectors conducted a telephonic exit meeting with Mr. Jeff Reinhart, Site Vice President, and other members of your staff regarding unresolved item 2007007-07. The licensee acknowledged the findings during the meeting.

On April 7, 2011, the inspectors presented the inspection results to Mr. J. Reinhart, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION
KEY POINTS OF CONTACT

Licensee Personnel

R. Acker, Licensing Engineer
H. Faulhaber, Division Manager, Nuclear Contruction and Projects
M. Ferm, Manager, Systems Engineering
M. Frans, Manager, Engineering Programs
J. Goddell, Division Manager, Nuclear Performance Improvement and Support
D. Guinn, Supervisor, Regulatory Compliance
R. Haug, Manager, Training
J. Herman, Division Manager, Nuclear Engineering
T. Nellenbach, Division manager, Nuclear Operations
J. Reinhart, Site Vice President
M. Smith, Manager, Operations
T. Uehling, Manager, Chemistry
J. Wiegand, Supervisor, Engineering Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000285/2010-006-00	LER	Reactor Trip Due to Erroneous Moisture Separator Trip Signal (Section 4OA3)
05000285/2011-001-00	LER	Inadequate Flooding Protection Due To Ineffective Oversight (Section 4OA3)

Opened and Closed

05000285/2011002-01	NCV	Inadequate Operating Instruction Results in a Loss of Auxiliary Feedwater (Section 1R20)
05000285/2011002-02	NCV	Failure to Determine the Cause of the Out Of Tolerance Condition Regarding Reactor Protection System (Section 1R22)
05000285/2011002-03	NCV	Failure to Submit a Timely Licensee Event Report (Section 1R22)
05000285/2011002-04	NCV	Failure to Verify Design Adequacy of Refueling Water Tank Vortex Eliminator (Section 4OA3)
05000285/2010-005-01	LER	Inoperability of the Emergency Diesel Generator Fuel Oil Transfer System (Section 4OA3)

Closed

05000285/2007007-07	URI	Safety Injection Refueling Water Tank Vortexing (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	DATE
OI-EW-1	Extreme Weather	18
OI-EW-1, Attachment 1A	Extreme Weather	November 16, 2010

Section 1R04: Equipment Alignment

CONDITION REPORTS

2011-2480

WORK ORDERS

282634-02 325856-04 346655-13

PROCEDURES

NUMBER	TITLE	REVISION
FC-1145	Scaffold Request Form	16
OI-CS-1	Containment Spray - Normal Operation	38
OI-EE-3	125 VDC System Normal Operation	20
OI-FP-1	Fire Protection System Water System	76
PED-CSS-12	Standard Specification for Scaffold Construction	5
SO-M-35	Scaffolding Installation Control	19

DRAWINGS

NUMBER	TITLE	REVISION
11405-E-120	120/208 Volt AC and 125 Volt DC Local Power Panels, Sheets 31-34, 37, 49, 59-60,67-70	
11405-M-266	Fire Protection Flow Diagram, Sheet 1B	29
E-23866-210-130	Composite Flow Diagram, Safety Injection and Containment Spray System P&ID, Sheet COV	61
E-23866-210-130	Safety Injection and Containment Spray System Flow Diagram P&ID, Sheet 1	106

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
SDBD-CS-131	Containment Spray	31
SDBD-EE-202	DC Distribution	18
SDBD-FP-115	Fire Protection	31
TS 2.3	Fort Calhoun Technical Specification 2.3, “Emergency Core Cooling System”	221
TS 2.7	Fort Calhoun Technical Specification 2.7, “Electrical Systems”	264

Section 1R05: Fire Protection

PROCEDURES

NUMBER	TITLE	REVISION
GO-G-102	Standing Order, Fire Protection Program Plan	9
SO-G-103	Standing Order, Fire Protection Operability Criteria and Surveillance Requirements	25
SO-G-28	Standing Order, Station Fire Plan	79
SO-G-58	Standing Order, Control of Fire Protection System Impairments	37
SO-G-91	Standing Order, Control and Transportation of Combustible Materials	26

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
EA-FC-97-001	Fire hazards Analysis Manual	15
FC05814	UFHA Combustible Loading Calculation	11
USAR 9.11	Updated Safety Analysis Report, Fire Protection Systems	21

Section 1R07: Heat Sink Performance

CONDITION REPORTS

2010-1139	2010-2417	2010-3899	2010-5894

WORK ORDERS

386035

PROCEDURES

NUMBER	TITLE	REVISION
PE-RR-CCW-0100	Disassembly, Cleaning, and Repair of CCW Heat Exchanger – Raw Water Side	36

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
TS 2.4	Fort Calhoun Technical Specification 2.4, “Containment Cooling”	249

Section 1R11: Licensed Operator Requalification

PROCEDURES

NUMBER	TITLE	REVISION
AOP-19	Loss of Shutdown Cooling	16
LOR TPMP	Licensed Operator Requal Training Program Master Plan	40
OPD-3-11	Licensed Activation and Watch station Maintenance	16
SO-G-26	Training and Qualification Programs Standing Orders	56

Section 1R12: Maintenance Effectiveness

CONDITION REPORTS

2010-2303 2010-5481 2010-5489 2010-5520

PROCEDURES

NUMBER	TITLE	REVISION
PBD-16	Program Basis Document, Maintenance Rule	8
PED-SEI-34	Maintenance Rule Program	8

MISCELLANEOUS DOCUMENTS

TITLE	REVISION / DATE
Maintenance Rule Scoping Data Sheet CRFILT	2a
Status of Equipment in MR Category (a)(1) or (a)(1) review	January 7, 2011

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

CONDITION REPORTS

2011-0820	2011-2083	2011-2404	2011-2600	2011-2643
2011-2686				

PROCEDURES

NUMBER	TITLE	REVISION
SO-G-87	Non-Routine Activities Requiring Formalized Plans	13
SO-M-100	Standing Order, Conduct of Maintenance	54
SO-M-101	Standing Order, Maintenance Work Control	89

MISCELLANEOUS DOCUMENTS

TITLE	DATE
Summary of Scheduled Activities Affecting Plant Risk	Week of 3/13/11
Summary of Scheduled Activities Affecting Plant Risk	Week of 3/20/11
Summary of Scheduled Activities Affecting Plant Risk	Week of 1/17/11

Section 1R15: Operability Evaluations

CONDITION REPORTS

2011-1121	2011-1278	2011-1304	2011-1308	2011-1941
2011-1942	2011-2246			

WORK ORDERS

387293	387797	405269	405575

PROCEDURES

NUMBER	TITLE	REVISION / DATE
IC-CP-07-001	Calibration of Pressure Gauges	13

PROCEDURES

NUMBER	TITLE	REVISION / DATE
OP-ST-AFW-3009	Auxiliary Feedwater Pump FW-6, Recirculation Valve and Check Valve Tests	19

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISON
	ASME OMb Code-2000 Addenda to ASME OM CODE – 1998 Code for Operation and Maintenance of Nuclear Power Plants	2000
FC-1401	Reportability Evaluation Checklist 2011-1941	
FC-1401	Reportability Evaluation Checklist 2011-1942	
NOD-QP-31.1	Operability Evaluation Form 2011-1941	
NOD-QP-31.1	Operability Evaluation Form 2011-1942	

Section 1R18: Plant Modifications

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
EC 51773	Electrically Remove S1, S2, and S3 VA-64A-COMP High Winding Temperature Sensors From the Protected Circuitry	March 2, 2011
HCOM-SB-20	Trane Air Conditioning: Operation and Troubleshooting, Robertshaw MP13, MP23 and MC20 Solid State Motor Protectors	July 1, 1981

Section 1R19: Postmaintenance Testing

CONDITION REPORTS

2011-0330 2011-0334 2011-0336

WORK ORDERS

402378

PROCEDURES

NUMBER	TITLE	REVISION / DATE
OI-CC-1	Component Cooling System Normal Operation	67
OP-ST-AFW-3011	Auxiliary Feedwater Pump FW-10, Steam Isolation Valve, and Check Valve Tests	10
OP-ST-ESF-0022	S1-2 Automatic Load Sequencer Test	29
OP-ST-RPS-0008	Reactor Manual Trip Test	February 5, 2011
SE-ST-CCW-3002	CCW Pump Base Line Curve Procedure	10

DRAWINGS

NUMBER	TITLE	REVISION
11405-M-10	Auxiliary Coolant Component Cooling System	29

Section 1R20: Refueling and Other Outage Activities

CONDITION REPORTS

2011-0836 2011-0839

PROCEDURES

NUMBER	TITLE	REVISION
EOP-00	Standard Post Trip Actions	27
EOP-01	Reactor Trip Recovery	13
OI-AFW-4	Auxiliary Feedwater System Startup and System Operation	78
OP-2A	Plant Startup	102
OP-4	Load Change and Normal Power Operation	44

Section 1R22: Surveillance Testing

CONDITION REPORTS

2011-0266 2011-0272 2011-0554 2010-6135

WORK ORDERS

402266 384204

PROCEDURES

NUMBER	TITLE	REVISION / DATE
IC-ST-IA-3009	Operability Test of IA-YCV-1045-C and Close Stroke Test of YCV-1045	January 21, 2011
IC-ST-RPS-0042	Quarterly Functional Test of RPS Trip Logic	5
IC-ST-RPS-0044	Quarterly Functional Test of Steam Generator Low Pressure and Asymmetric Steam Generator Transient RPS Bi-stable Trip Units	6
OP-ST-AE-0001	Personnel Access Lock (Pal) O-Ring Seal Test	21
OP-ST-AFW-3011	Auxiliary Feedwater Pump FW-10, Steam Isolation Valve and Check Valve Tests	10

DRAWINGS

NUMBER	TITLE	REVISION
11405-M-252	Steam P&ID	45
11405-M-253	Feedwater and Blowdown P&ID	47
C-4175 Sh4	Control Valve Air Source Valve Lineup/Listing I&C Equipment List	19
E-23866-411-003	Reactor Protection System Functional Diagram	10
E-23866-411-061	TMLP Function and Wiring Diagram	15

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
5.19	Technical Specifications – Containment Leakage Rate Testing Program	259

Section 1EP6: Drill Evaluation

PROCEDURES

NUMBER	TITLE	REVISION
TBD-EPIP-OSC-1A	Recognition Category A - Abnormal Rad Levels/Radiological Effluent	1

PROCEDURES

NUMBER	TITLE	REVISION
TBD-EPIP-OSC-1F	Recognition Category F - Fission Product Barrier Degradation	1
TBD-EPIP-OSC-1H	Recognition Category H - Hazards and Other Conditions Affecting Plant Safety	1
TBD-EPIP-OSC-1S	Recognition Category S - System Malfunction	1

Section 4OA1: Performance Indicator Verification

PROCEDURES

NUMBER	TITLE	REVISION
OP-ST-RC-3001	Reactor Coolant System (RCS) Leak Rate Test	34

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Various Operator Logs	January 1, 2010 through December 31/2010
FCSG-47	RCS Leak Rate Monitoring Program	0
NEI 99-02	Regulatory Assessment Indicator Guideline	6

Section 4OA2: Identification and Resolution of Problems

CONDITION REPORTS

200503188	200503188	200602289	200602297	200602439
200700204	200701228	20073312	2008-0032	2008-1984
2009-1439	2009-2466	2009-2468	2009-5373	2009-5925
2010-0631	2010-1759	2010-2060	2010-2253	2010-2341
2010-2452	2010-2650	2010-2910	2010-3290	2010-3337
2010-3350	2010-3368	2010-3558	2010-3995	2010-4089
2010-4100	2010-4617	2010-5193	2010-5760	2010-6077
2010-6199	2010-6457	2010-6769	2010-6835	2011-0053
2011-0090	2011-1542			

WORK REQUESTS

145111	145419	145420	145421	147059
149705	149819	150395	150819	151035

151240	152094	152821	152951	152992
153139	153587	153814	153852	154532
154708	156073	156075	156751	156752
156753	157619	157823	157824	158057
156753	157619	157823	157824	158057
158217	158267	158614	159090	159194
159566	159625			

WORK ORDERS

232289	287130	287130	346569	347195
367507	368019	373900	377701	378503
378505	378790	380661	381224	381745
383201	385061	385102	385171	387348
388988	391900	391901	391902	396755
398955	400299			

PROCEDURES

NUMBER	TITLE	REVISION
ARP-CB-1,2,3/A6	Annunciator Response Procedure	42
FCSG-45	Operator Challenge Program	4
PBD-10	Boric Acid Corrosion Control	13

ENGINEERING CHANGES (EC)

NUMBER	TITLE	REVISION
32387	Turbine Control System Upgrade	0
33719	F/G1-Rec Generator ST-2 Frequency Recorder	1
35741	Travelling Screens Replacement	1
44046	Replace FC-2817 thru FC-2824	0
44892	Upgrade Valve Trim and Operator for FCV-1101 and FCV-1102	0
45882	Revise Mounting of DCS GPS Antenna for Better Satellite Reception	0
48179	Upgrade BAST Area Heating	0
49474	Replacement for PIA-207	0
50011	Replace UR-6684 and UR-6683. No Replacement Recorders Available	0

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
	Quality Control Inspection Report	20090258
	Quality Control Inspection Report	20090381
	Quality Control Inspection Report	20090333
	System Training Manual Vol. 37 Reactor Coolant System	42
13038	Primary Coolant Pump	6
FC-1389	Boric Acid Corrosion Control Screening Form for Valves/Threaded Connections/Bolted Connection/ Piping/Instrumentation and Control 2009-5373	

Section 4OA3: Event Follow-Up

CALCULATIONS

NUMBER	TITLE	REVISION / DATE
FAI/07-66	Evaluation of the Fort Calhoun SIRWT Capabilities for Vortex Elimination in the ECCS Suction Piping	June, 2007
FAI/07-80	Evaluation of the Fort Calhoun SIRWT Potential for Vortexing During a Draindown Transient	July, 2007
FAI/07-125	Fort Calhoun Transient Tests Investigating the Potential for Vortex Formation in the SIRWT Suction Flow	October, 2007
FC05455	ECCS Pump NPSH and 383 Series Valve Stroke Times	4
FC05598	Verification of SIRWT RAS Initiation Switch Setpoints	1