UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

November 14, 2006

R. T. Ridenoure
Vice President
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION - NRC INTEGRATED INSPECTION REPORT 05000285/2006004

Dear Mr. Ridenoure:

On September 30, 2006, 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 October 6, 2006, with Mr. Jeff Reinhart, Site Director, 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 four NRC-identified findings and one self-revealing finding of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. Additionally, a 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 non-cited violations (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violations or significance of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; 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 facility.

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

Sincerely,

/RA/

Zachary K. Dunham, Chief
Project Branch E
Division of Reactor Projects

Docket: 50-285
License: DPR-40

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RIV:RI:DRP/E	SRI:DRP/E	C:DRS/EB1	C:DRS/OB	
LMWilloughby	JDHanna	JAClark	RLNease	
T-ZKDunham	***T-ZKDunham***	***/RA/***	***/RA/***	
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C:DRS/EB2	STA:DRS	C:DRP/E	
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285
License:	DPR-40
Report:	05000285/2006004
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	Fort Calhoun Station FC-2-4 Adm. P.O. Box 399, Highway 75 - North of Fort Calhoun Fort Calhoun, Nebraska
Dates:	July 1 through September 30, 2006
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Approved By:	Zachary K. Dunham, Chief, Project Branch E Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 0500285/2006004; 7/1/2006 - 9/30/2006; Fort Calhoun Station; Permanent Plant Modifications, Refueling and Other Outage Activities, Access Control to Radiologically Significant Areas, Other Activities.

The report covered a 3-month period of inspections by resident inspectors and announced inspections by a health physicist, a senior engineering reactor inspector, engineering reactor inspectors, engineering contractors, a senior operations engineer, an operations engineer and a senior emergency preparedness inspector. Five Green findings, all of which were noncited violations, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." 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 3, dated July 2000.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a Green, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to use the correct total dead weight of the replacement pressurizer in two design calculations.

  The failure to correctly translate the total dead weight of the replacement pressurizer into design calculations is a performance deficiency because the licensee failed to meet 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and the cause was reasonably within the licensee's ability to foresee and correct. The finding is more than minor because it affects the design control attribute of the initiating events objective listed in Manual Chapter 0612, "Power Reactor Inspection Reports," Appendix B. Because the incorrect weight was used in the analyses, the analyses were re-evaluated. Since the finding did not result in a loss of function or mitigation capability, the violation has very low safety significance (Green), using Manual Chapter 0609, "Significance Determination Process."

  This finding has a crosscutting aspect in the area of human performance because the licensee failed to use conservative assumptions in their decision-making. This caused the licensee to miss opportunities to revise specific design documentation for the pressurizer. A contributing factor is the licensee's regard toward the replacement pressurizer as a "like-for-like" replacement for the original pressurizer. Although the design function of the replacement pressurizer is similar to the original pressurizer, specific design parameters, such as weight, volume, and heater capacity, are actually different (Section 1R17).

Cornerstone: Mitigating Systems

- Green. A noncited violation was identified for failure to comply with Technical Specification 2.1.1.(3), which required two operable decay heat removal loops. This failure resulted in a condition where only one shutdown cooling train was operable. This condition existed for 2 days before being detected by operations personnel.

  This finding was determined to be greater than minor in that it affected the "Configuration Control" attribute of the Mitigating Systems cornerstone. The inspectors evaluated this finding using Manual Chapter 0609, Appendix G, because the condition occurred and was identified during shutdown conditions. Using Checklist 2, the inspectors determined that the finding screened as Green because the condition did not increase the likelihood that a loss of decay heat removal would occur due to failure of the system itself. This condition was entered into the licensee's corrective action program as Condition Report 200603965. This finding has a crosscutting aspect in the area of human performance associated with decision making because operations personnel incorrectly concluded that the shutdown cooling header was operable (Section 1R20).

- Green. The inspectors identified a noncited violation of Technical Specification 5.8.1.c for failure to have an adequate procedure to implement postfire safe shutdown actions. Specifically, Procedure SO-G-28, "Station Fire Plan," Revision 61, Attachment 14, failed to list operable diagnostic instrumentation, actions needed to respond to faults on 4 kV busses, and had operators re-enter an area without ensuring it was safe to enter.

  This finding is of greater than minor safety significance because it had the potential to impact the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. Consequently, the inspectors evaluated these deficiencies using Manual Chapter 0609, Appendix F. Since the issue involved postfire safe shutdown actions in the auxiliary building related to maintaining reactor coolant system inventory and maintaining a heat sink, had existed for more than 30 days, and had a moderate degradation rating, the issue did not screen out in Phase 1. Because of the room volumes and the forced ventilation flow rates, the sources did not generate sufficient heat in the hot gas layer to damage the targets. Consequently, in accordance with the Appendix F, Step 2.3, of the Phase 2 significance determination process, the inspectors concluded that this finding was of very low safety significance. In addition, this finding had a crosscutting aspect in the area of human performance because the licensee did not ensure complete, accurate and up-to-date procedures needed to implement manual actions existed for postfire safe shutdown (Section 4OA5.3).

- Green. The inspectors identified a noncited violation of Technical Specification 5.8.1.c for failure to have an adequate procedure to implement postfire safe shutdown actions. Specifically, simulated operator actions during a walkthrough of Procedure AOP-06, “Fire Emergency,” could not be performed in the time specified in engineering calculations, nor were all appropriate steps specified.

This finding is of greater than minor safety significance because it had the potential to impact the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. Specifically, the issue involved postfire safe shutdown actions in the auxiliary building upon evacuation from the control room related to maintaining a heat sink. Because of other actions that would likely have been taken, the inspectors concluded this issue had a low degradation rating and, therefore, the inspectors concluded the issue was of very low safety significance in Phase 1. In addition, this finding had a crosscutting aspect in the area of human performance because the licensee did not ensure complete, accurate and up-to-date procedures needed to implement the actions existed (Section 4OA5.4).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors reviewed two examples of a self-revealing, noncited violation of Technical Specification 5.11.1 in which workers failed to obtain high radiation area access authorization and associated radiological briefing before entering the area. The first example occurred on March 26, 2005, when a worker received a dose rate alarm while assisting with the movement of an equipment cutter known to generate a high radiation area. The second example occurred on September 16, 2006, when a worker received two dose rate alarms while working on two fire detectors in the overhead. The worker passed through a high radiation area while performing work on the second fire detector. For the first example, the licensee enhanced pre-job briefings to verify appropriate authorizations and briefings via self and peer checking. For the second example, corrective actions are still being implemented.

  This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure/contamination control) and affects the Occupational Radiation Safety cornerstone objective, in that the failure to obtain high radiation area authorized access and associated radiological briefings resulted in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an ALARA finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess doses. Additionally, this finding had a cross-cutting aspect in the area of human performance because the workers failed to use error prevention tools such as self and peer checking. (Section 2OS1)

B. Licensee Identified Findings

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. These violations and corrective action tracking numbers (condition report numbers) are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The unit began this inspection period in Mode 1 at full rated thermal power and operated at 100 percent until August 18, 2006, when power was decreased on the unit to 97 percent to perform Moderator Temperature Coefficient testing. On August 20, reactor power was increased to 100 percent, where the plant remained until September 9. On September 9 the unit was manually tripped in order to start the refueling outage for replacement of the steam generators, pressurizer and reactor vessel head components. The unit remained shutdown at the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's implementation of changes to the facility structures, systems, and components; risk-significant normal and emergency operating procedures; test programs; and the updated final safety analysis report in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments." The inspectors utilized Inspection Procedure 71111.02, "Evaluation of Changes, Tests, or Experiments," for this inspection.

The procedure specifies five as the minimum sample size of safety evaluations and a combination of 10 applicability determinations and screenings, with the emphasis on screenings.

The inspectors reviewed five safety evaluations performed by the licensee since the last NRC inspection of this area at Fort Calhoun Station, with an emphasis on replacement nuclear steam supply system components. The evaluations were reviewed to verify that licensee personnel had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. The inspectors reviewed 20 licensee-performed applicability determinations and screenings in which, licensee personnel determined that neither screenings nor evaluations were required to ensure that the exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59. Procedures, evaluations, screenings, and applicability determinations reviewed are listed in the attachment to this report

The inspectors reviewed and evaluated a sample of recent licensee condition reports to determine whether the licensee had identified problems related to the 10 CFR 50.59 evaluations, entered them into the corrective action program, and resolved technical concerns and regulatory requirements.

The inspection procedure specifies inspectors' review of a required minimum sample of 5 licensee safety evaluations and 10 applicability determinations and screenings (combined). The inspectors completed review of 5 licensee safety evaluations and 20 applicability determinations and screenings (combined).

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Equipment Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the three risk important systems listed below and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walkdown to the licensee's Updated Safety Analysis Report (USAR) and Corrective Action Program to ensure problems were being identified and corrected.

- July 18, 2006, Raw Water to Component Cooling Water Heat Exchangers AC-1B, AC-1C, and AC-1D while AC-1A was out of service for maintenance on relief valve RW-221

- July 25, 2006, Component Cooling Water system that supports Spent Fuel Pool Cooling

- September 22, 2006, Spent Fuel Pool cooling system with the fuel from the core fully offloaded

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors walked down the six plant areas listed below to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the USAR to determine if the licensee identified and corrected fire protection problems.

- July 17, 2006, Gas Decay Tank WD-29C vault, Room 17 (Fire Area 6.1)
- July 25, 2006, Cask Decontamination Area, Room 67 (Fire Area 20.7)
- July 25, 2006, Auxiliary Building 1025 Elevation Work Area, Room 71 (Fire Area 28)
- July 29, 2006, Review of effect of underground fire main break on other portions of the plant
- August 24, 2006, Spent Resin Storage Tank Room (Fire Areas 20.1 and 20.6)
- September 29, 2006, Upper Level of Auxiliary Building, Room 69 (Fire Area 20.7)

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 Semi-annual Internal Flooding

a. Inspection Scope

The inspectors: (1) reviewed the USAR, the flooding analysis, and plant procedures to assess seasonal susceptibilities involving internal flooding; (2) reviewed the Corrective Action Program to determine if the licensee identified and corrected flooding problems; (3) inspected underground bunkers/manholes to verify the adequacy of (a) sump pumps, (b) level alarm circuits, (c) cable splices subject to submergence, and (d) drainage for bunkers/manholes; (4) verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and (5) walked down the areas listed below to verify the adequacy of: (a) equipment seals located below the flood line,

(b) floor and wall penetration seals, (c) watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

- September 29, 2006, Auxiliary Building 971 Elevation (Rooms 21 and 22)

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspection Activities

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. On August 1, 2006 the inspectors observed training scenarios that involved various equipment failures. The first scenario included a main feed water line rupture while the second scenario included a primary to secondary leak with a station blackout. The inspectors compared performance in the simulator with performance observed in the control room during this inspection period. The focus of the inspection was on high-risk licensed operator actions, operator activities associated with the emergency plan, and previous lessons-learned items. These items were evaluated to ensure that operator performance was consistent with protection of the reactor core during postulated accidents.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Regional Biennial Examination

a. Inspection Scope

This inspection was held during the last week of the biennial examination testing cycle, which ended the week of August 7, 2007. The inspectors reviewed the overall pass/fail results of the individual job performance measure operating tests, simulator operating tests, and written examinations administered by the licensee during the operator licensing requalification cycles and biennial examination. Ten separate crews participated in simulator operating tests, and job performance measure operating tests, totaling 46 licensed operators. While there were a few individual job performance measure failures, all of the licensed operators tested passed the biennial examination.

During the inspection, the inspectors reviewed and observed biennial examination simulator job performance measures, in-plant job performance measures, the simulator static exam, written examination, licensed operator classroom instruction, and the plant control room crew. They also reviewed a sample of licensed operator annual medical forms and procedures governing the medical examination process.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the two maintenance activities listed below in order to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule, 10 CFR Part 50 Appendix B, and the Technical Specifications.

- September 25, 2006, Instrument Air Dryer failures
- September 28, 2006, Fuel Oil Tank FO-38 Level Switch LS-2120

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

Risk Assessment and Management of Risk

a. Inspection Scope

The inspectors reviewed the five assessment activities listed below to verify: (1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognizes, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) the licensee identified and corrected problems related to maintenance risk assessments.

- July 11, 2006, Equipment stored on top of containment

- July 17, 2006, water supply from Blair, Nebraska out of service resulting in Condensate Storage Tank level lowering to less than 67 percent

- September 7, 2006, review of licensee's risk assessment for the Fall 2006 refueling outage and replacement of major components to ensure shutdown risk management objectives were acceptable (e.g. reduced inventory considerations, control of heavy loads, alternate power)

- September 10, 2006, Component Cooling Water Pump AC-3B out of service with the reactor on shut down cooling and 161kV off-side power unavailable

- September 12, 2006, Component Cooling Water Pump AC-3B out of service with the reactor at midloop conditions

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plants status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the USAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any Technical Specifications; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- July 19, 2006, Diesel Generator 2 Jacket Water Temperature High and Lube Oil Cooler Temperature High alarms while the machine was loaded for monthly surveillance test

- August 30, 2006, YCV-817B Diesel Generator 2 Room Fresh Air Supply Damper lower two damper vanes secured closed by grout

- September 29, 2006, Containment Duct Relief Port open to atmosphere

Documents reviewed by the inspectors included: CR 200603052, CR 200603597, and CR 200604230.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B)

a. Inspection Scope

The inspection procedure requires inspection of a minimum sample size of five permanent plant modifications.

The inspectors reviewed eight permanent plant modification packages and associated documentation, such as; implementation reviews, safety evaluation applicability determinations, and screenings, to verify that they were performed in accordance with regulatory requirements and plant procedures. The inspectors also reviewed the procedures governing plant modifications to evaluate the effectiveness of the program for implementing modifications to risk-significant systems, structures, and components, such that these changes did not adversely affect the design and licensing basis of the facility. Procedures and permanent plant modifications reviewed are listed in the attachment to this report. Further, the inspectors interviewed certain of the cognizant design and system engineers for the identified modifications as to their understanding of the modification packages and process.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems concerning the performance of permanent plant modifications by reviewing a sample of related condition reports. The reviewed condition reports are identified in the attachment.

The inspection procedure specifies inspectors' review of a required minimum sample of five permanent plant modifications. The inspectors completed review of eight permanent plant modifications.

b. Findings

Failure to Translate Replacement Pressurizer Weight Into Design Calculations

Introduction. The inspectors identified a Green, NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to use the correct total dead weight of the replacement pressurizer in two design calculations. In addition, this finding has a human performance crosscutting aspect.

Description. On August 8, 2006, the inspectors reviewed Engineering Change EC 32447, "Pressurizer Replacement." Engineering Change EC 32447, Section 4.3.3, states design loads of the replacement pressurizer for the structural analysis will be a total dead weight consisting of the replacement pressurizer filled with cold water including insulation. This weight is about 191 kips. The inspectors identified that in two calculations, FC 03122, "10-inch Surge Line Break," and FC 07085, "Pressurizer Anchor Bolts", Fort Calhoun Station personnel used a replacement pressurizer weight that is substantially lower than the pressurizer total dead weight, as stated in Engineering Change EC 32447. Calculation FC 03122, the referenced loading analysis for the slab carrying the replacement pressurizer, used a total weight of 181 kips. Calculation FC07085, the referenced seismic analysis for the pressurizer anchoring, used a total weight of 144 kips.

After discussion with licensee personnel, the analyses were reevaluated using more conservative weight assumptions. The issue was entered into the corrective action program as CR 200603413.

Analysis. The failure to correctly translate the total dead weight of the replacement pressurizer into design calculations is a performance deficiency because the licensee failed to meet 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and the cause was reasonably within the licensee's ability to foresee and correct. The finding is more than minor because it affects the design control attribute of the initiating events cornerstone objectives listed in Manual Chapter 0612, "Power Reactor Inspection Reports," Appendix B. Because the incorrect weight was used in the analyses, the analyses were re-evaluated. Since the finding did not result in a loss of function or mitigation capability, the violation has very low safety significance (Green), using Phase 1 of Manual Chapter 0609, "Significance Determination Process."

This finding has a crosscutting aspect in the area of human performance because the licensee failed to use conservative assumptions in their decision-making. This caused the licensee to miss opportunities to revise specific design documentation for the pressurizer. A contributing factor is the licensee's regard towards the replacement pressurizer as a "like-for-like" replacement for the original pressurizer. Although the design function of the replacement pressurizer is similar to the original pressurizer, specific design parameters, such as weight, volume, and heater capacity, are actually different.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, states, in part, measures shall be established to assure that applicable regulatory requirements and the design basis, for structures, systems, and components, are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to this, as of August 8, 2006, Fort Calhoun Station personnel had failed to correctly translate the replacement pressurizer total dead weight into two analysis: (1) seismic design of pressurizer anchor bolts; and (2) integrity of the slab and compartment supporting the pressurizer.

Because this failure to comply with 10 CFR Part 50, Appendix B, Criterion III, is of very low safety significance and has been entered into the licensee's corrective action program as CR 200603413, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000285/2006004-01 Failure to Translate Replacement Pressurizer Weight Into Design Calculations.)

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the five postmaintenance test activities listed below of risk significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly re-aligned, and deficiencies during testing were documented. The inspectors also reviewed the USAR to determine if the licensee identified and corrected problems related to postmaintenance testing.

- September 6, 2006, Replace Filter or Regulator Assembly for IA-HCV-2883B-FR (Work Order 00217639-01)

- September 6, 2006, In-office review of post maintenance test on Charging Pump CH-1A following performance of SP-CP-08-480-1B3A, “Calibration of Protective Relays for 480-1B3A Bus,” Revision 14

- September 6, 2006, Replace Steam Generator RC-2A Blow-down to Blow-down Tank FW-7 Control Valve HCV-1390 (Work Order 00218435-01)

- September 6, 2006, repair the Fire Main Rupture between FP-106 and FP-104 (Work Order 00244394-01)

- September 6, 2006, in-office review of postmaintenance test on High Pressure Safety Injection Pump SI-2C following performance of SP-CP-08-480-1B3A, “Calibration of Protective Relays for 480-1B3A Bus,” Revision 14

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan, compliance with the Technical Specifications, and adherence to commitments in response to

Generic Letter 88-17, “Loss of Decay Heat Removal”: (1) the risk control plan; (2) tagging/clearance activities; (3) reactor coolant system instrumentation; (4) electrical power; (5) decay heat removal; (6) spent fuel pool cooling; (7) inventory control; (8) reactivity control; (9) containment closure; (10) reduced inventory or midloop conditions; (11) refueling activities; (12) cooldown activities; and (13) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities. Due to the licensee’s refueling outage continuing past the end of the inspection period, activities such as heatup and restart were not yet inspected. The inspectors' reviews particularly focused on establishment of plant conditions necessary for the replacement of the major components (i.e., steam generators, pressurizer, reactor vessel head). Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

Introduction. The inspectors identified a Green NCV for failure to comply with Technical Specification 2.1.1.(3), which required two operable decay heat removal loops. This failure resulted in a condition where only one shutdown cooling train was operable. This condition existed for 2 days before being detected by operations personnel.

Description. On September 9, 2006, the licensee commenced shutdown of the plant in support of the Fall 2006 refueling outage. On September 10, at approximately 9:30 a.m., operations personnel performed the initial valve lineup per OI-SC-1, “Shutdown Cooling Initiation,” Revision 42, for establishment of shutdown cooling. (This procedure established the configuration of systems necessary to further lower plant temperature and maintain core cooling.) At 12:30 p.m., reactor coolant temperature decreased to less than 210°F and pressure was lowered below the necessary minimum for single reactor coolant pump operation. Once this condition existed, Technical Specification 2.1.1.(3) became applicable and the steam generators became unavailable as a heat removal source due to inability to run reactor coolant pumps to dissipate decay heat.

On September 12, at approximately 7:30 p.m., a valve lineup was subsequently performed for the purpose of re-verifying the configuration of the system. Operators performing this valve lineup discovered that manual isolation Valve SI-173 (Shutdown Heat Exchanger AC-4A & 4B Outlet Cross Connect Valve) was locked shut. The valve was immediately restored to the open position. The licensee determined that, on September 9, 2006, when Procedure OI-SC-1 had last been performed, a procedure requirement to open Valve SI-173 had been inadvertently signed as completed without the valve actually being repositioned.

The inspectors determined that, had a failure of the operating Train A of shutdown cooling occurred, Train B would not have been available. Significant diagnosis would have been required during a postulated event in order to determine the cause of lack of flow. Further, licensee Procedure AOP-19, “Loss of Shutdown Cooling,” Revision 12, which the operators would use to respond to such an event, did not require them to either verify or reposition Valve SI-173. The initial determination by operations personnel (i.e., that Train B of shutdown cooling had been operable while in the isolated condition) was questioned by the inspectors. Fort Calhoun Station's operability determination of the shutdown cooling train was later revised to reflect that it had in fact been inoperable.

Analysis. The inspectors determined that the failure to comply with Technical Specifications for the reactor coolant system was a performance deficiency. This finding was determined to be greater than minor in that it affected the "Configuration Control" attribute of the Mitigating Systems cornerstone. The inspectors evaluated this finding using Manual Chapter 0609, Appendix G, because the condition occurred and was identified during shutdown conditions. Using Checklist 2 the inspectors determined that the finding screened as Green because the condition did not increase the likelihood that a loss of decay heat removal would occur due to failure of the system itself. This finding has a crosscutting aspect in the area of human performance associated with decision making because operations personnel incorrectly concluded that the shutdown cooling header was operable.

Enforcement. Technical Specification 2.1.1.(3) requires, in part, that with "T_{cold} less than 210ºF with fuel in the reactor and all reactor vessel head closure bolts fully tightened, at least two of the decay heat removal loops . . . shall be operable." Operable is defined in the Technical Specifications as "when it is capable of performing its specified function(s)." Contrary to the above, on September 10-12, 2006, only one train of shutdown cooling was operable. This violation of Technical Specification 2.1.1.(3) is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (NCV 05000285/2006004-02). This violation was entered into the licensee corrective action program as CR 200603965.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the USAR, procedure requirements, and Technical Specifications to ensure that the five surveillance activities listed below demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms set points. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- July 27, 2006, observed the Independent Spent Fuel Storage Facility surveillance test MSLT-DSC-TriVis, "Helium Mass Spectrometer Leak Test Procedure" Revision FtC-0

- August 16, 2006, Surveillance Test IC-ST-MS-0031, “Channel Calibration of Steam Generator RC-2B Channel B Pressure Loop B/P-905,” Revision 14

- August 18, 2006, review of the leak detection activities conducted in accordance with OP-ST-RC-3001, “Reactor Coolant System Leak Rate Test,” during a period of slightly elevated leakage

- August 23, 2006, Surveillance Test IC-ST-RPS–0055, “Calibration of Power Range Safety Channel C,” Revision 2

- August 29, 2006, In service Test SE-ST-MS-3005, “Main Steam Safety Valves Set pressure Using Trevitest Equipment,” Revision 4

Documents reviewed by the inspectors are shown above.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspectors performed in-office reviews of revisions to the Fort Calhoun Station Emergency Plan, including Revision 13 to Section D, Revision 33 to Section H, and Revision 19 to Section J. The inspectors also reviewed Revisions 40 and 41 to Emergency Plan Implementing Procedure OSC-1, "Emergency Classification." The revisions were submitted between April and August, 2006. The revisions (1) added procedural direction for implementation of the requirements of 10 CFR Part 72 for a dry fuel storage program, (2) added new emergency action level (7.1) for damage to a loaded dry fuel cask confinement boundary, (3) revised protective action recommendation guidance to specify the criteria for a sheltering recommendation in lieu of an evacuation recommendation during short term (< 1 hour) radiological releases with limited dose projections, and (4) relocated one emergency alert siren a minor distance with the concurrence of the Department of Homeland Security.

The revisions were compared to their previous revisions, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, to the criteria of NEI 99-01, "Methodology for Development of Emergency Action Levels," Revision 2, and to the standards in 10 CFR 50.47(b) to determine if the revisions were adequately conducted following the requirements of 10 CFR 50.54(q). This review was not documented in a Safety Evaluation Report and did not constitute approval of licensee changes, therefore, these revisions are subject to future inspection.

The inspectors completed one sample during the inspection.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas (HRAs), and worker adherence to these controls. The inspectors used the requirements in 10 CFR Part 20, the Technical Specifications, and the licensee's procedures required by Technical Specifications as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone

- Controls (surveys, posting, and barricades) of radiation, high radiation, and potential airborne radioactivity areas in the Reactor, Spent Fuel, and Auxiliary Buildings

- Radiation work permits, procedures, engineering controls, and air sampler locations

- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms.

- Barrier integrity and performance of engineering controls in two potential airborne radioactivity areas

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem Committed Effective Dose Equivalent

- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within the spent fuel pool.

- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection

- Corrective action documents related to access controls

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls such as, required surveys, radiation protection job coverage, and contamination controls during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

The inspectors completed 20 of the required 21 samples.

b. Findings

Introduction. The inspectors reviewed two examples of a self-revealing, noncited violation of Technical Specification 5.11.1, in which workers failed to obtain a high radiation area access authorization and associated radiological briefing before entering into the area. The violation had very low safety significance.

Description. The first example occurred on March 26, 2005, when a worker received a dose rate alarm while participating in the movement of equipment cutters with radiation readings greater than 100 millirem per hour at 30 centimeters. An investigation into the dose rate alarm revealed the individual was briefed and authorized for work activities, which did not include entries into high radiation areas. The individual voluntarily assisted another work group with the cutter movement but did not consider the limitations of his prior briefing and the high radiation area access authorization. In addition, the radiation protection technician covering the work activity assumed all individuals in the work area were appropriately briefed and authorized for the work activity. The licensee enhanced pre-job briefings to include additional radiation protection staff and worker self and peer checking to verify appropriate authorizations and briefings were performed.

The second example occurred on September 16, 2006, when a worker received two dose rate alarms while working on two fire detectors in the overhead between the equipment hatch and the pressurizer cubicle. The work scope was discussed with radiation protection personnel at the containment control point but was not sufficiently communicated with the radiation protection technician providing the pre-job surveys. This led the radiation protection technician to only survey and evaluate the fire detector that was in an open area and not the second area. After completing work on the fire detector in the open area, the worker used the nearby cable trays to gain access to the second fire detector where he passed in close proximity to the safety injection line. The worker received two dose rate alarms (going to and returning from) the second fire detector. The worker then exited containment and reported the alarms to radiation protection. The worker's dose rate alarm was set at 40 millirem per hour, the peak dose rate seen by the electronic alarming dosimeter was 102 millirem per hour, and a survey of the safety injection line after the event identified 110 millirem per hour at 30 cm. The worker failed to obtain radiological conditions and access authorization for the second area entered.

Analysis. The failure to obtain high radiation area access authorization and associated radiological briefings before entering the area is a performance deficiency. This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure/contamination control) and affects the Occupational Radiation Safety cornerstone objective, in that the failure to obtain high radiation area authorized access and associated radiological briefings resulted in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an ALARA finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess doses. Additionally, this finding had a crosscutting aspect in the area of human performance because the workers failed to use error prevention tools such as self and peer checking.

Enforcement. Technical Specification 5.11.1 states, in part, that in lieu of the "control device" required by 10 CFR 20.1601(a) and 20.1601(c), each high radiation area, as defined in 10 CFR 20.1601, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto controlled by a Radiation Work Permit. Any individuals permitted to enter such areas shall be provided with a continuously integrating and alarming radiation-monitoring device and may enter after the dose rate levels in the area have been established and personnel are made knowledgeable of them. Contrary to Technical Specifications, workers entered high radiation areas without obtaining the required radiological briefing and were not specifically authorized to enter the areas. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (Condition Reports CR 200501675 and CR 200604123), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/2006004-03, Failure to obtain high radiation area access authorization and associated radiological briefing.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors used the requirements in 10 CFR Part 20 and the licensee's procedures required by Technical Specifications as criteria for determining compliance. The inspectors interviewed licensee personnel and reviewed:

- Three outage work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures

- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups

- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents

- Exposure tracking system

- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding

- Workers use of the low dose waiting areas

- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner

- Specific sources identified by the licensee for exposure reduction actions and priorities established for these actions, and results achieved against since the last refueling cycle

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

- Corrective action documents related to the ALARA program and follow-up activities such as initial problem identification, characterization, and tracking

The inspectors completed 4 of the required 15 samples and 7 of the optional samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

The inspectors reviewed licensee documents from January 1, 2005, through June 30, 2006. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee's technical specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 3, were used to verify the basis in reporting for each data element.

The inspectors completed the required sample (1) in this cornerstone.

Public Radiation Safety Cornerstone

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

The inspectors reviewed licensee documents from January 1, 2005, through June 30, 2006. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded PI thresholds and those reported to the NRC. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the PI data. PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 3, were used to verify the basis in reporting for each data element.

The inspectors completed the required sample (1) in this cornerstone.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Fire Protection Unresolved Item Review

a. Inspection Scope

As part of the unresolved item closeout inspection, the inspectors assessed: (1) the corrective actions implemented for each specific unresolved item, (2) the self assessment performed to evaluate the fire protection program progress and readiness for this inspection, (3) plans implemented related to manual actions for 10 CFR Part 50, Appendix R, Section III.G.2 areas.

The inspectors conducted this inspection through documentation review and interviews with engineering and licensing personnel.

b. Observations and Findings

The inspectors noted that the licensee had taken significant steps to identify the extent of condition related to the unresolved items identified in the August 2005 triennial fire protection inspection. However, the inspectors noted that the licensee had not completed their procedure revisions at the time of this inspection. Similarly, the licensee had not finalized the engineering review of the engineered safety feature actuations.

The self assessment performed in June 2006 provided critical recommendations of the fire protection organization's progress related to the unresolved items and the level of detail in the plan to resolve the large number of manual actions for Appendix R, Section III.G.2 areas that did not have exemptions in place. For example, the self-assessment noted that the plans for resolving the use of manual actions, as documented in CR 200601090 did not have sufficient detail to drive the issue to resolution.

.2 Problem Identification and Resolution for Radiation Protection

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings

No findings of significance were identified.

.3 Routine Review of Identification and Resolution of Problems with a Operator Work Around

a. Inspection Scope

The inspectors chose one issue (one inspection sample) for more in-depth review to verify that the licensee personnel had taken corrective actions commensurate with the significance of the issue. The inspectors reviewed the corrective actions associated with this condition including the licensee's classification of the issue being an operator work around. The inspectors also performed a review of operator workarounds, control room deficiencies, and control room burden lists. The inspectors focused on the cumulative effects of the workaround on the reliability/availability of mitigating systems and the corresponding impact on operators to respond in a correct and timely manner to plant transients and accidents. The inspectors reviewed the deficiencies against the licensee's Procedure OPD-4-17, "Control Room Deficiencies, Operator Burdens, and Operator Workaround," Revision 16, that described the programs for handling workarounds and deficiencies. The following issue was evaluated:

- Review of CR 2005005837 Degraded FI-417, Flow Indicator for Cooling Water Flow from VA-1B

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000285/2005008-01: Failure to maintain the safety injection and refueling water tank valves free of fire damage

Introduction. The inspectors determined that the failure to have the cable separation required by 10 CFR Part 50, Appendix R, Section III.G.2, to the suction valves located between the safety injection and refueling water tank and the safety injection pumps would not have resulted in closure of the valves. The short that could result would not generate sufficient voltage to actuate the solenoid for the suction valves. This failure to comply with 10 CFR Part 50, Appendix R, Section III.G.2 constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy.

Description. During the triennial fire protection inspection in August 2005, the team determined that a fire in Fire Area 20 could potentially cause loss of redundant trains of systems and equipment credited in the postfire safe shutdown analysis. Specifically, the safe shutdown analysis credited the use of Safety Injection Pumps SI-2A or SI-2B taking suction from the safety injection and refueling water tank.

The team had determined that: (1) the postfire safe shutdown analysis credited Valves LCV-383-1 and LCV-383-2 for the safety injection system to accomplish its shutdown function and at least one of the two valves must remain free of fire damage; (2) a single hot short on Cable EB3884 (Valve LCV-383-1) or Cable EA3890 (Valve LCV-383-2) could cause the associated valve to fail in the undesired (closed) position; and (3) the licensee had routed both cables in cable trays that are located less that 10 feet apart horizontally. The licensee initiated CR 200504001 to place this item into their corrective action program and had established an hourly fire watch for this fire area as an interim compensatory measure.

During this inspection, the inspectors: (1) reviewed Operability Evaluation for Valves LCV-383-1 and LCV-383-2, (2) verified that the indicating lamp had a 2000-ohm resistor, (3) verified that the solenoid had a maximum resistance of 885 ohms, and (4) verified the solenoid required 90 Vdc to actuate. The worst-case scenario resulted from a short from the close circuit to the solenoid actuation circuit that placed the indicating lamp and solenoid in series in the 125 Vdc circuit. Analyzing the circuit determined that the solenoid would draw 38.4 Vdc, which would not actuate the solenoid and inadvertently close the valves.

Analysis. Routing the cables for safety-related valves needed for postfire safe shutdown within 10 feet of each other was a performance deficiency for failure to meet the separation requirements specified in 10 CFR Part 50, Appendix R, Section III.G.2. This finding was determined to be of minor safety significance because it would not have impacted the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. Specifically, a fire in Fire Area 20 did not have the potential to cause damage to circuits that could adversely affect the ability of the licensee to provide makeup to the reactor coolant system via the safety injection and refueling water tank.

Enforcement. This failure to comply with 10 CFR Part 50, Appendix R, Section III.G.2 constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee entered this deficiency into their corrective action program as CR 200504001. The inspectors determined that the licensee had initiated Project Number FC 38203 in April 2006 to route one of the cables in a conduit or relocate to another fire area because of the continued noncompliance with 10 CFR Part 50, Appendix R, Section III.G.2.

.2 (Closed) Unresolved Item 05000285/2005008-02: Lack of an evaluation of fire-induced automatic actuation signals on a fire area basis

Introduction. The inspectors determined that the failure to evaluate fire-induced actuations of engineered safety feature actuation system sensors and cables as required by 10 CFR Part 50, Appendix R, Section III.G.2 would not have resulted in actuation of components needed for hot shutdown. The evaluation that was performed did identify circuits subject to spurious actuation needed for cold shutdown, which could be repaired within the 72 hours allowed. This failure to comply with 10 CFR Part 50, Appendix R, Section III.G.2 constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy.

Description. During the triennial fire protection inspection in August 2005, the team determined that the safe shutdown analysis had not evaluated engineered safety feature actuation system automatic control systems or related instrumentation and cables that could have a significant impact on safety if damaged during a fire. For example, for Fire Area 20 the safe shutdown analysis credits the use of safety injection pumps taking suction from the safety injection and refueling water tank. However, if a recirculation actuation signal occurred because of fire damage, the discharge valves for the tank would close and the suction for the pumps could be transferred to a dry containment sump, which could damage the pumps. The licensee entered this finding into the corrective action program as CR 200503738 and established an hourly fire watch for this fire area as an interim compensatory measure.

During this inspection, the inspectors reviewed Calculation EA 06-008, "Engineered Safety Features Actuation System (ESFAS) Fire-Induced Failure Evaluation," Revision 0, and discussed the results with the fire protection engineer. Calculation EA 06-008 evaluated the circuits related to the re-circulation actuation signal, the containment spray actuation signal, the safety injection actuation signal, the containment isolation actuation signal, and the steam generator isolation signal. The inspectors determined that the evaluation appropriately identified each sensor and sensor cable for faults. The evaluation identified that many circuits needed for cold shutdown would require manual actions to resolve spurious operation and made corrective action recommendations. Some conclusions did not clearly indicate that the spurious operation would not affect achieving hot shutdown.

Consequently, the inspectors interviewed the fire protection engineer and reviewed Calculation EA-FC-89-055, "10 CFR Part 50, Appendix R, Safe Shutdown Analyses," Revision 12. This review confirmed that components affected were not required for a long period, were needed to achieve cold shutdown, and were being addressed in the update to Procedure AOP-06, "Fire Emergency," Revision 16. Consequently, the inspectors concluded that the potential circuit failures would have little effect on the ability of the licensee staff to achieve hot shutdown.

Analysis. The failure to evaluate engineered safety feature actuation systems for fire-induced circuit failures resulted in a performance deficiency for failure to meet the separation requirements specified in 10 CFR Part 50, Appendix R, Section III.G.2. This finding was determined to be of minor safety significance because it would not have impacted the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. Specifically, the failure to evaluate fire-induced actuations, including the impact on safe shutdown, of the engineered safety feature actuation systems instrumentation and cables did not affect response activities to achieve hot shutdown.

Enforcement. This failure to comply with 10 CFR Part 50, Appendix R, Section III.G.2 constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee entered this deficiency into their corrective action program as CR 200503738. At the time of this inspection, the licensee had recently received the evaluation from their contractor and had not completed all of their engineering reviews.

.3 (Closed) Unresolved Item 05000285/2005008-03: Inadequate procedure for implementing the fire protection program as required by Technical Specification 5.8.1.c.

Introduction. The inspectors identified a Green NCV of Technical Specification 5.8.1.c for failure to have an adequate procedure to implement postfire safe shutdown actions. Specifically, Procedure SO-G-28, "Station Fire Plan," Revision 61, did not provide adequate instructions for operators to mitigate the effects of fire damage.

Description. During the triennial fire protection inspection in August 2005, the team identified several deficiencies related to the postfire safe shutdown procedures. Operators used Procedure AOP-06, "Fire Emergency," Revision 11 to implement the detailed response when evacuating the control room, including manual actions. Procedure SO-G-28 provided instructions for operators to mitigate the effects of fire damage to safe shutdown equipment in plant areas other than the control room and the cable spreading room. Procedure SO-G-28, Attachment 14, "Restoration of Safe Shutdown Conditions in the Event of a Fire," described the fire areas that required the use of manual operator actions to mitigate fires in those areas for fires other than a control room evacuation.

As a result of tabletop walkthroughs and simulator evaluations using Procedures AOP-06 and SO-G-28, the team had determined that Procedure SO-G-28: (1) was not referred to in Procedure AOP-06; (2) did not direct operators to enter Attachment 14 nor did operators refer to the attachment; (3) did not identify the diagnostic instrumentation that may be relied upon for a fire in each fire area; (4) main body did not provide operators detailed information identifying the manual actions to be performed in response to a fire; (5) did not provide operators information as to which, if any, manual actions are time critical; and (6) for Fire Area 43, required operators to re-enter the area if a fire had occurred to close Manual Valve IA-3119. In summary, the team concluded that manual actions were not reliable and feasible because of the lack of diagnostic instruments being identified, the poor coordination among the various procedures, and operator's lack of familiarity with Procedure SO-G-28, Attachment 14, which identified key manual actions needed.

During this inspection, the inspectors identified postfire safe shutdown components in Fire Areas 20, 32 and 43 which required manipulation to safely shutdown the reactor for fires outside the control room. For Fire Area 20 (Room 69), the inspectors concluded that Procedure SO-G-28 provided appropriate guidance through redirection to AOP-32, "Loss of 4160 Volt or 480 Volt Bus Power," Revision 10, and EOP-20,"Functional Recovery Procedure," Revision 18. The third action in this fire area involved valving in raw water to the control room HVAC upon loss of normal cooling water. The inspectors considered this action low risk since the control room heat-up would be gradual. However, the inspectors noted that, at the time of this finding, the procedure remained deficient in that it had not identified the instruments that remained operable.

For Fire Area 32 (Room 19), Procedure SO-G-28, Attachment 14 failed to list operable diagnostic instrumentation and actions needed to respond to spurious operation of components powered from the 4 kV busses. Similarly, for Fire Area 43 (Room 81), Procedure SO-G-28, Attachment 14, failed to identify operable diagnostic instruments and required operators to re-enter the room when it may not have been habitable. The inspectors determined that the references to other emergency and abnormal operating procedures provided appropriate implementing instructions.

The licensee had entered these deficiencies into their corrective action program as CRs 200503731, 200504006, and 200504203. The inspectors verified that the licensee had revised Procedure SO-G-28 to refer to Attachment 14 and to include the operable diagnostic information in Attachment 14. In addition, the licensee had initiated revisions to Procedure AOP-06 to incorporate the guidelines contained in Procedure SO-G-28 and provided more detailed mitigation steps. Upon final approval all guidance would be contained in Procedure AOP-06. This finding had a cross-cutting aspect in the area of human performance because the licensee did not ensure complete, accurate and up-to-date procedures needed to implement manual actions for postfire safe shutdown.

Analysis. The failure of Procedure SO-G-28 to provide adequate instructions to operators to perform manual actions to mitigate the consequences of fire damage and ensure hot shutdown could be achieved was a performance deficiency for failure to meet Technical Specification 5.8.1.c. Specifically, Procedure SO-G-28, Attachment 14, failed to list operable diagnostic instrumentation, actions needed to respond to faults on 4 kV busses, and had operators re-enter an area without knowing it would be safe. This deficiency was more than minor in that it had the potential to impact the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. Consequently, the inspectors evaluated these deficiencies using Manual Chapter 0609, Appendix F

The actions for Fire Area 32 (Room 19) were postfire safe shutdown functions in the auxiliary building related to maintaining reactor coolant system inventory (inadvertent operation of the power-operated relief valves), had existed for more than 30 days, and had a moderate degradation rating. Consequently, the issue did not screen out in Phase 1. During the Phase 2 evaluation, the inspectors identified the ignition sources (air compressor motor, air compressor oil, turbine-driven auxiliary feedwater pump oil, electrical control cabinet for the air compressor, motor driven auxiliary feedwater pump motor) and the targets (thermoset cable). One component, compressor electrical cabinets, did not screen out and required use of the NUREG-1805 model for a room with forced ventilation to determine the hot gas layer temperature. Because of the room volume and the forced ventilation flow rate, the electrical cabinet did not generate sufficient heat in the hot gas layer to damage the thermoset cables.

The actions for Fire Area 43 (Room 81) were postfire safe shutdown functions in the auxiliary building related to maintaining a heat sink (operability of auxiliary feedwater), had existed for more than 30 days, and had a moderate degradation rating. Consequently, the issue did not screen out in Phase 1. During the Phase 2 evaluation, the inspectors identified the ignition sources (ventilation unit motors and wood staged in a metal gang box) and the targets as the E/P converter for the auxiliary feedwater air-operated valves and the electric panels for the main steam code safeties. One component, electric cables to the E/P converter for the air-operated auxiliary feedwater valve, did not screen out and required use of the NUREG-1805 model for a room with forced ventilation to determine the hot gas layer temperature. Because of the room volume and the forced ventilation flow rate, the wood in the metal gang box (assumed the wood was not enclosed) did not generate sufficient heat in the hot gas layer to damage the cables to the E/P converter.

However, because the potential for fire damage did not exist in Fire Areas 32 and 43 as determined by the Appendix F, Step 2.3 Phase 2 significance determination process for each fire area, the inspectors concluded that this finding was of very low safety significance (Green).

Enforcement. Technical Specification 5.8.1.c. requires that written procedures and administrative policies shall be established, implemented and maintained covering fire protection program implementation. Procedure SO-G-28 provided the guidance to operators, including manual actions, to achieve postfire safe shutdown. Inspection Procedure 71111.05T, Enclosure 2, specified the criteria that must be met for manual actions to be considered feasible without an approved exemption to 10 CFR Part 50, Appendix R. Contrary to the above, the inspectors determined that Procedure SO-G-28 failed to meet the following manual action feasibility criteria: (1) procedure guidance failed to identify exactly what manual actions were needed, (2) diagnostic instruments that remained operable for a fire in each fire area were not identified, and (3) directed operators to the area without any guidelines for when it would be safe to manipulate a component in the same area. Because this finding is of very low safety significance and has been entered into the corrective action program (CR 200504203), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/2006004-04, Failure to implement reasonable and feasible manual actions.

.4 (Closed) Unresolved Item 05000285/2005008-04: Inadequate fire safe shutdown procedure for control room evacuation

Introduction. The inspectors identified a Green NCV of Technical Specification 5.8.1.c for failure to have an adequate procedure to implement postfire safe shutdown actions. Specifically, simulated operator actions during a walkthrough of Procedure AOP-06, "Fire Emergency," Revision 12, could not be performed in the time specified in engineering calculations nor were all appropriate steps specified.

Description. During the triennial fire protection inspection in August 2005, the team identified, during timed walkthroughs of AOP-06, Section II, "Control Room Evacuation," that the procedure had inadequate guidance. The team determined that Procedure AOP-06, Section II: (1) identified establishing control for alternate shutdown at AI-179, Auxiliary Feedwater Panel, and AI-185, Alternate Shutdown Panel, (2) failed to identify a time frame for establishing auxiliary feedwater whereas calculations specified time frames as short as 12 minutes, and (3) prior to establishing control at Panel AI-179, required the communicator to manually throttle Valves HCV-1107B, "Steam Generator RC-2A Auxiliary Feedwater Inlet Valve," and HCV-1108B, "Steam Generator RC-2B Auxiliary Feedwater Inlet Valve," to 75 percent closed.

Further, the team determined that: (1) the communicator can easily meet the time line in the calculations with the valves in their normally closed position. However, if the valves receive a spurious open signal prior to throttling, interviews with operators indicated that the valves may not be able to be manually throttled, and (2) Procedure AOP-06, Section II, identified no contingency actions to throttle the valves closed or for establishing control at Panel AI-179 if the valves were not throttled closed.

During this inspection, the inspectors verified the licensee had corrected the deficiencies identified by the team. Further, the licensee entered this finding into the corrective action program as CR 200503731 and revised Procedure AOP-06 to include contingency actions should the valves open prior to completion of manual throttling. This finding had a crosscutting aspect in the area of human performance because the licensee did not ensure complete, accurate and up-to-date procedures needed to implement the actions.

<u>Analysis</u>. The failure of Procedure AOP-06 to provide sufficient guidance was a performance deficiency for failure to meet Technical Specification 5.8.1.c. Specifically, the procedure failed to ensure that response personnel had the appropriate guidance and equipment to allow them to carry out the functions of limiting auxiliary feedwater flow to the steam generators when needed. This deficiency was more than minor in that it had the potential to impact the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. Consequently, the inspectors evaluated these deficiencies using Manual Chapter 0609, Appendix F.

Because of other actions that would, likely, have been taken, the inspectors concluded this issue had a low degradation rating and, therefore, the inspector concluded the issue had very low safety significance in the Phase 1 evaluation.

<u>Enforcement</u>. Technical Specification 5.8.1.c. requires that written procedures and administrative policies shall be established, implemented and maintained covering fire protection program implementation. Procedure AOP-06, Section II, provided the guidance to operators, including manual actions, to achieve postfire safe shutdown for a control room evacuation. Inspection Procedure 71111.05T, Enclosure 2, specified the criteria that must be met for manual actions to be considered feasible without an approved exemption to 10 CFR Part 50, Appendix R. Contrary to the above, the inspectors determined that Procedure AOP-06, Section II, failed to ensure that manual operation of auxiliary feedwater valves would be accomplished prior to the times specified in engineering calculations and failed to ensure sufficient guidance and tools existed for equipment operators to accomplish the task. Specifically, the procedure specified no time limit, and the communicator, during timing evolutions, indicated that if the valves were open the 12-minute time limit would not be met and he had no way of informing the control room supervisor because he did not carry a radio. Because this finding is of very low safety significance and has been entered into the corrective action program (CR 200503731), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/2006004-05, Inadequate alternate shutdown procedure.

.5 <u>(Closed) LER 05000285/2006002-00</u>, Inadequate Design Control Results in Potentially Insufficient Auxiliary Feedwater Flow

The details of this condition are discussed in Section 4OA7 of this report. This LER is closed.

4OA6 <u>Meetings</u>

<u>Exit Meeting Summary</u>

The inspectors discussed the preliminary results of the fire protection unresolved item review with Mr. J. Reinhart, Site Director, and other members of licensee management on July 21, 2006. The inspectors returned proprietary information examined during the inspection to the licensee. The inspectors conducted a telephonic exit meeting with Mr. Joe McManis, Manager, Nuclear Licensing, and other licensee personnel on August 18, 2006. Licensee management acknowledged the inspection results.

On August 10, 2006, the operator licensing inspectors conducted a debrief meeting to present the licensed operator requalification inspection results to the Licensee's management team. During the debrief, the inspectors informed the management team they had obtained permission to retain copies of six medical certification forms containing privacy information act material. It had also been agreed this material would be shredded upon issuance of the inspection report. The licensee was informed that a final exit for the inspection would be conducted after the requalification program was completed and the NRC had reviewed the final results. On September 20, 2006, a final exit, which described the inspection results, was conducted by the inspectors via telephone with Mr. D. Weaver, Supervisor of Operations Training. The licensee acknowledged the findings presented in both the briefing and the final exit meeting. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On August 11, 2006, the inspectors presented the safety evaluation and permanent plant modifications inspection results to Mr. J. Reinhert, Site Director, and other members of the staff who acknowledged the findings. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

On August 30, 2006, the inspectors presented the results of the emergency plan change inspection to Mr. C. Simmons, Supervisor, Emergency Preparedness. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

On September 22, 2006, the inspectors presented the occupational radiation safety inspection results to Mr. J. Reinhart, Site Director, and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

The results of the resident inspector activities were presented to Mr. J. Reinhart, Site director, and other members of licensee management on October 6, 2006. The inspectors confirmed that proprietary information examined during the inspection period was returned to the licensee. Licensee management acknowledged the inspection findings.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- Title 10 CFR Part 50, Appendix B, Section III, “Design Control,” states, in part, that “Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety related functions of the SSCs.” Contrary to the above, the electrical power supply to flow transmitter FT-1368 (Motor Driven Auxiliary Feedwater Pump Suction Flow Transmitter) was not safety-related. During an event the flow transmitter and associated recirculation valve may not perform its design function consequently challenging the ability of the Motor Driven Auxiliary Feedwater Pump to provide cooling to the steam generators. This finding only had very low safety significance because it was a design or qualification deficiency confirmed not to result in loss of operability. This finding was identified in the licensee’s corrective action program as CR 200602855 and was reported as LER 05000285/2006-002-00.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bannister, Plant Manager
B. Blessie, Supervisor, Operations Engineer
D. Buell, Fire Protection Engineer
T. Byrne, Licensing Engineer (Title 10 CFR 50.59 Program Coordinator)
G. Cavanaugh, Supervisor, Regulatory Compliance
S. Cofaul, ALARA Technician, Radiation protection
M. Core, Manager, System Engineering
H. Faulhaber, Division Manager, Engineering
M. Ferm, Manager, Shift Operations
W. Goddell, Nuclear Training Manager
D. Guinn, Licensing Engineer
W. Hansher, Lead, Nuclear Safety Review
R. Haug, manager, Radiation Protection
K. Hyde, Supervisor, mechanical Engineering
R. Jaworski, Licensing Engineer
G. Labs, Simulator Supervisor
D. Lakin, Manager, Corrective Action Program
T. Maine, Supervisor, Radiation Protection
E. Matzke, Compliance Engineer
J. McManis, Manager, Licensing
T. Nellenbach, Manager, Operations
M. Pohl, Principal Reactor Engineer, Operations
M. Quinn, Nuclear Engineering and Computing Projects Supervisor
J. Reinhart, Site Director
R. Short, Manager, NSSS Replacement Components
C. Simmons, Supervisor, Emergency Preparedness
M. Tesar, Division manager, Nuclear Support Services
J. Tills, Manager, Maintenance
D. Travsch, Manager, Quality
D. Weaver, Operations and Technical Training Supervisor
J. Willett, Principle Reactor Engineer Fuels, Operations
C. Williams, Supervisor, Radiation Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open and Closed

05000258/2006004-01	NCV	Failure to Translate Replacement Pressurizer Weight Into Design Calculations (Section 1R17)
05000285/2006004-02	NCV	Failure to Maintain Shutdown Cooling Train Operable as Required by Technical Specification 2.1.1.(3) (Section 1R20)

05000285/2006004-03	NCV	Failure to Obtain High Radiation Area Access Authorization and an Associated Radiological Briefing (Section 2OS1)
05000285/2006004-04	NCV	Failure to Implement Reasonable and Feasible Manual Actions (Section 4OA5.3)
05000285/2006004-05	NCV	Inadequate Alternate Shutdown Procedure (Section 4OA5.4)

Closed

05000285/2005008-01	URI	Failure to Maintain the Safety Injection and Refueling Water Tank Valves Free of Fire Damage (Section 4OA5.1)
05000285/2005008-02	URI	Lack of an Evaluation of Fire-Induced Automatic Actuation Signals on a Fire Area Basis (Section 4OA5.2)
05000285/2005008-03	URI	Inadequate Procedure for Implementing the Fire Protection Program as Required by Technical Specification 5.8.1.c. (Section 4OA5.3)
05000285/2005008-04	URI	Inadequate Fire Safe Shutdown Procedure for Control Room Evacuation (Section 4OA5.4)
05000285/2006002-00	LER	Inadequate Design Control Results in Potentially Insufficient Auxiliary Feedwater Flow (Section 4OA7)

LIST OF DOCUMENTS REVIEWED

Section 1R02: Evaluations of Changes, Tests, or Experiments

10 CFR 50.59 Evaluations

FC-071145, LTR-RCPL-04-75, OPPD Replacement Pressurizer
EC 33109
EC 38303
FC-154B for EC-31589
FC-154B for EC-38331
10 CFR 50.59 Screenings

EC 33116
FC-154A, EC-33105
EC 33117
EC 33109
EC-154A for EC-31589 (RSG)
FC-154A for EC-31589 (RSG Type C-6 Nozzle Dams)
FC-154A for EC-33106
EC 33153
EC 25764 for USAR Section 14 Revision
EC 33104

Applicability Determinations

FC-68C for EC 33105
EC 33116
EC 33117
EC 33109
EC 33115
FC-68C for EC 31589
FC-68C for EC 33106
EC 33153
EC 25764 for USAR Section 14 Revision
EC 33104

Procedures

NOD-QP-3, "10 CFR 50.59 and 10 CFR 72.48 Reviews"

Section 1RO4: Equipment Alignment

Licensee Procedure OI-SFP-1, "Spent Fuel Pool Cooling Normal Operations," Revision 29

Licensee Procedure ARP-CB-1,2,3/A1, "Annunciator Response Procedure A1 Control Room Annunciator A1", Revision 26

Drawing 11405-M11, "Auxiliary Coolant Spent Fuel Pool Cooling System Flow Diagram P&ID," Revision 52

Section 1RO5: Fire Protection

Standing Order SO-G-28, "Station Fire Plan," Revision 66

Standing Order SO-G-102, "Fire Protection Program," Revision 7

Abnormal Operating Procedure AOP-6, "Fire Emergency," Revision 17

USAR, Section 9.11, "Fire Protection Systems"

Section 1RO6: Flood Protection Measures

Probabilistic Risk Assessment Summary Notebook, Revision 4

Individual Plant Examination Submittal, dated December 1993

Section 1R11: Licensed Operator Requalification Program

Open Simulator Discrepancy Reports (All)
Closed Simulator Discrepancy Reports Summary from January 2006 thru May 2006
Simulator Configuration Review Group (SCRG) meeting minutes for 2005
Simulator Annual Performance Test book for 2006

Simulator Steady State Testing Packages for 100% and 30% Power
Simulator Transient Testing Packages for Tests Three, Eight, and Ten
Current Simulator Differences List
Core physics testing packages for simulator, Cycle 23.
Low Power Physics Test data from the plant, Cycle 23.
Simulator Modification Procedures
Verification and Validation Procedures
Operator licensing tracking system active operator licenses (R4 OLTS report)
Current operator license list from Fort Calhoun Station
AP 21-001, Conduct of Operations, Rev. 35
AI 21-100, Operations Guidance and Expectations, Rev. 6
AI 30B-005, Conduct of Simulator Activities for Licensed Operator Training, Rev.8A
AP 30B-001, Licensed Operator Requalification Training Program, Rev. 7A
AP 30B-006, Shift Engineer/Shift Technical Advisor Requalification Training Program, Rev. 3
DTI 204, Operator Requalification JPM Preparation, Validation, and Administration

Section 1R12: Maintenance Effectiveness

Condition Reports

200503725	200505469	200600189	200601570
200603628			

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

Standing Order SO-O-21, “Shutdown Operations Protection Plan,” Revision 25

Condition Report 200602982

Control Room Operating Logs, dated July 16 and July 17, 2006

Risk evaluation and risk management actions per e-mail from John Fluehr, OPPD dated July 18, 2006

Section 1R17B: Permanent Plant Modifications

Plant Modifications

Number	Title	Revision
EC 32447	Replacement Pressurizer	0
EC 33105	Pressurizer Replacement	0
EC 33106	Steam Generator Large Bore Piping	0
EC 33116	Pressurizer Heater Cable Replacement	0
EC 33109	Containment Opening	0
EC 31589	Fort Calhoun - Replacement Steam Generators (Component)	0

EC 33153	Fort Calhoun - Replacement Reactor Vessel Head (Component)	0
EC 33104	Steam Generator Replacement	0

Engineering Changes

Number	Title	Revision
EC 38331	Safety Injection Phase Performance for Safety Injection and Containment Spray Systems Calculation No. FC07077	0
EC 33115	Temporary Transformer/RC-3A Tie-In	0
EC 33117	Replacement Pressurizer Instrument Modification	0
EC 38303	Recirculation Phase System Performance for Safety Injection and Containment Spray Systems	0

Drawings

ISO WD-2072, Sh.1	File 8939	9
ISO CH-2049, Sh. 1	File 8187	9
04-30991-01	Y-Globe Valve, Socket Ends...Size 2, Class 1878	0
11405-S-39	Reactor Plant Ground Floor Plan El. 1013'-0" Reinf. Sh.1	5

Calculations

FC 03122	10" Surge Line Break Effect on Pressurizer Slab and Walls below Pressurizer Compartment	1
FC 07085	Pressurizer Anchor Bolts	0
FC07172 (Bechtel Calculation 25036-C-029)	Evaluation of Containment Structure for Construction Opening	0
Combustion Engineering Calculation 0-SEC-15	Determination of Pressurizer Heater Capacity	7/12/67
FC 06974 (Areva Calculation) 32-5046461-00	FCS RSG – Decay Heat Removal Cap. In Nat. Circ. Analysis	4/1/04
32-5046526-00	FCS RSG – Loss of Load to Both Steam Generators Analysis	10/22/04

FC 07186	Fort Calhoun Scaling Calculation for Replacement Pressurizer Level Transmitters	3
CN-RVHP-05-59	Fort Calhoun Head Lift NUREG-0612 Evaluation	1
WB-CN-ENG-05-32	Fort Calhoun - Cap Screw Design	1
FC 03231	FCS RCS Support Validation	0

Procedures

Number	Title	Revision
SO-G-21	Standing Order Modification Control	78
PED-GEI-3	Preparation of Modification	42
PED-QP-2	Configuration Change Control	29
PSC Procedure F&Q 15.0	Precision Surveillance Corporation Field and Quality Control Procedure for Tendon Re-stressing	1
PSC Procedure F&Q 15.2	Precision Surveillance Corporation Field and Quality Control Procedure for Bearing Plate Concrete Inspection	0

Miscellaneous Documents

Number	Title	Revision
NPM-210	Nuclear Procurement Manual	13
N/A	Licensing Amendment Request Status Log	15
SA-06-23	Self Assessment Report, 10CFR50.50 Implementation	7/27/06
N/A	Watlow Pressurizer Heater Accelerated Life Test Status Report	7/12/06
FCSG-23	10 CFR 50.59 Resource Manual	5
FC-07145, LTR-RCPL-05-115	Final Design Licensing Report for the OPPD Replacement Pressurizer	0
FCP-KBS-05-00014	Accelerated Life Test Procedure for Heaters of RPZR	1
FCP-KBS-06-0002	RPZR Heater Accelerated Life Test Results for Short Term Electrical Failures	0
LIC-05-0107	Fort Calhoun Station Unit No. 1 License Amendment Request, “Updated Safety Analysis Report Revision for Radiological Consequences Analysis for Replacement NSSS Components”	10/31/05

NUREG 0800	Standard Review Plan for the Review of Safety Reports for Nuclear Power Plants	2
AREVA Engineering Information Record	FCS RSG - Control System Evaluation, 51-5050728-01	1
EA-FC -02-028	Appendix K Power Uprate Evaluation, Section 5	0
Email from Alan Wang (NRC) to Leonard M. Willoughby (NRC)	AST Accident Dose - Criteria for Categorical Exclusion	8/10/06
LTR-RCPL-05-135	Final Design Licensing Report for the OPPD Replacement Reactor Vessel Head and Rapid Refueling Package (RRVH/RRP)	0
RFP 1758	Technical Specification for Design of Mirror Insulation for the Replacement Reactor Vessel Head for Omaha Public Power District, Fort Calhoun Station	0
MR FC-79-15	Replacement of Reactor Pressure Vessel and Seismic Skirt Insulation; Appendix 7.2, Section H, Contract 1318 Technical Specification	4/82

Condition Reports

CR 200603413	CR 200402963	CR 200504555	CR 200600896
CR 200600624	CR 00602152	CR 200601839	CR 200603179
CR 200504214	CR 200600395	CR 200603252	CR 200504503
CR 200402637	CR 200504503	CR 200500408	CR 200600750
CR 200602255	CR 200403490	CR 200601815	CR 200505022
CR 200600454	CR 200602693	CR 200603374	CR 200401985
CR 200503149	CR 00600195	CR 200501970	

Section 1R19: Postmaintenance Testing

Work Order 00217639-01, Replace Filter or Regulator Assembly for IA-HCV-2883B-FR

Procedure SP-CP-08-480-1B3A, “Calibration of Protective Relays for 480-1B3A Bus,” Revision 14

Work Order 00218435-01, Replace Steam Generator RC-2A Blow-down to Blow-down Tank FW-7 Control Valve HCV-1390

Work Order 00244394-01, Repair the Fire Main Rupture between FP-106 and FP-104

Section 1R20: Refueling and Other Outage Activities

Shutdown Safety Advisor’s Log dated September 13, 2006

Technical Specifications, Definitions Section, page 5

OI-SC-1, “Shutdown Cooling System,” Revision 42

Drawing D-4768, “Primary Plant Simplified Flowpath Diagram,” Revision 5

Abnormal Operating Procedure AOP-19, “Loss of Shutdown Cooling,” Revision 12

Root Cause Analysis Report for CR 200603965

Section 2OS1: Access Controls to Radiologically Significant Areas (71121.01)

Audits, Self-Assessments, and Surveillances

Quality Assurance Audit Report No. 49/58
Self-Assessment SA-06-02
Surveillance Report 58(3)-0506

Condition Reports

200500993, 200501625, 200501675, 200600870, 200601277, 20061866, 200603848, 200604123

Procedures

RP-202	Radiation Protection Radiological Surveys, Revision 26
RP-204	Radiological Area Controls, Revision 44
RP-208	Radiography, Revision 10
RP-602	Radiation Protection Personnel Dosimetry Issuance and Change-out, Revision 20
RP-608	Dose Calculations from Contamination, Revision 11
RPI-13	Radiological Posting Standards, Revision 2
SO-G-92	Conduct of Infrequently Performed Procedures, Revision 9
SO-G-101	Radiation Worker Practices, Revision 30
SO-O-47	Spent Fuel Pool Inventory Control, Revision 6

Radiation Work Permits

06-3001, 06-3520, 06-3533, and 06-3541

Sample Results and Surveys

Air Sample Form and Results for RWP 06-3541 on 09/21/06
Survey Numbers: 05-1173, 06-1088

Miscellaneous

2005 DAC-Hour Tracking Summary
Dose Rate Alarm Report
Shift Outage Manager's Reports

Section 2OS2: ALARA Planning and Controls (71121.02)

Audits, Self-Assessments, and Surveillances

Quality Assurance Audit Report No. 49/58
Self-Assessment SA-06-02
Surveillance Report 58(3)-0506

Condition Reports

200504826, 200505725, 200602354

Radiation Work Permits

06-3520, 06-3533, and 06-3541

Procedures

RP-301 ALARA Planning / RWP Development and Control, Revision 26

Miscellaneous

Shift Outage Manager's Reports

Section 4OA1: Performance Indicator Verification (71151)

Procedures

NOD-QP-40 NRC Performance Indicator Program, Revision 2

Miscellaneous

2005 Abnormal Batch Liquid and Gaseous Release Summary
2005 Batch Liquid and Gaseous Release Summary
2005 Liquid Effluents Continuous Mode
Surveillance Report Numbers: 63(3)-0606 and 63(3)-1105

Section 4OA5: Other Activities (71111.05T)

Procedures

AOP-06, “Fire Emergency,” Revisions 15 and 16
AOP-32, “Loss of 4160 Volt or 480 Volt Bus Power,” Revision 10
EOP-06, “Loss of All Feedwater,” Revision 12
EOP-20, “Functional Recovery Procedure,” Revision 18
FCSG, “Performing Risk Assessments,”
OPD-2-06, “Operations Department Duties and Responsibilities,” Revision 21
SO-G-28, “Station Fire Plan,” Revisions 61 and 65
SO-—100, “Conduct of Maintenance,” Revision 41
SO-O-1, “Conduct of Operations,” Revision 69

Drawings

11405-—253, “Flow Diagram, Steam Generator Feedwater and Blowdown,” Sheet 4, Revision 3

11405-S-64, “Auxiliary Building Sections,” Sheet 2, Revision 4

Calculations

EA 06-008, “Engineered Safety Features Actuation System (ESFAS) Fire-Induced Failure Evaluation,” Revision 0

EA-FC-89-055, “10 CFR Part 50, Appendix R, Safe Shutdown Analysis,” Revisions 11 and 12

EA-FC-97-001, “Fire Hazards Analysis (FHA) Manual,” Revision 11

EA-FC-97-044, “10 CFR Part 50, Appendix R, Cable Identification,” Revision 4

FC 05814, “UFHA Combustible Loading,” Revision 9

Condition Reports

200204316	200503731	200503738	200503750	200503979	200504001
200504006	200504203	200601090			

Miscellaneous

Engineering Information Record 51-9016709-00, “Fort Calhoun Station Transient Analysis, Manual Action Timeline and Feasibility Study,” dated June 21, 2006

Fisher-Rosemount Vendor Manual, “Type 657 Diaphragm Actuator, Sizes 30 - 70 and 87"

LIST OF ACRONYMS

CFR	*Code of Federal Regulations*
CR	Condition Report
NCV	noncited violation
NRC	Nuclear Regulatory Commission
SSC	Structure, System and Component
USAR	Updated Safety Analysis Report