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

August 12, 2008

D. J. Bannister
Vice President
Omaha Public Power District
Fort Calhoun Station FC-2-4
P.O. Box 550
Fort Calhoun, NE 68023-0550

Subject: FORT CALHOUN STATION NRC INTEGRATED INSPECTION REPORT 05000285/2008003

Dear Mr. Bannister:

On June 30, 2008, 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 July 9, 2008, with Mr. Rich Clemens, Division Manager Nuclear Engineering, 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 one self-revealing and four NRC identified findings of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. Additionally, three licensee-identified violations, which were determined to be of very low safety significance, are listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violations or the 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, 612 East Lamar Blvd, Suite 400, Arlington, Texas 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at the Fort Calhoun Station facility.

In accordance with 10 CFR 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/

Wayne C. Walker
Chief, Project Branch E
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/200803
W/Attachment: Supplemental Information

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SUNSI Review Completed __WCW__ ADAMS: √ Yes □ No Initials: __WCW__
√ Publicly Available □ Non-Publicly Available □ Sensitive √ Non-Sensitive

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RIV:RI:DRP/E	SRI:DRP/E	SPE:DRP/E	C:DRS/EB1	C:DRS/PSB
JCKirkland	JDHanna	GDReplogle	RLBywater	MPShannon
/RA/ T = JDHanna for	***/RA/ T-WCWalker for***	***/RA/***	***/RA/***	***/RA/***
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C:DRS/EB2	C:DRS/OB	C:DRS/PSB2	C:DRP/E	
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285
License:	DPR-40
Report:	05000285/2008003
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:	April 1 through June 30, 2008
Inspectors:	J. Hanna, Senior Resident Inspector J. Kirkland, Resident Inspector D. Stearns, Health Physics Inspector L. Ellershaw, Senior Reactor Inspector S. Makor, Reactor Inspector M. Young, Reactor Inspector P. Elkman, Senior Emergency Preparedness Inspector M. Haire, Senior Operations Engineer T. Pate, Reactor Inspector
Approved By:	Wayne C. Walker, Acting Chief, Project Branch E Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2008003; 04/01/2008 – 06/30/2008; Fort Calhoun Station, Integrated Resident and Regional Report, Inservice Inspection Activities, Access Control To Radiologically Significant Areas, ALARA Planning and Controls, Event Follow-up, Other Activities.

The report covered a 3-month period of inspection by resident inspectors and regional inspectors. Five Green noncited violations of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." 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. <u>NRC-Identified Findings and Self-Revealing Findings</u>

Cornerstone: Initiating Events

- <u>Green</u>. A self-revealing noncited violation of Technical Specification 5.8.1.a was identified for the failure to have an adequate procedure for plant cooldown. Specifically, on June 10, 2008, the plant cooldown procedure allowed the control room staff to unexpectedly draw an approximately 2700-gallon steam void in the reactor coolant system. The procedure failed to provide guidance to ensure the reactor vessel head and steam generator u-tubes were sufficiently cooled down before depressurizing the reactor coolant system. Contributors to the event included: 1) the failure to institutionalize related operating experience from NRC Generic Letter 81-21, "Natural Circulation Cooldown," dated May 5, 1981; and 2) the failure of plant operators to implement related training intended to avoid void formation. After voids formed, operators recognized the void indications, raised system pressure to collapse the steam voids, and then cooled the vessel head and steam generator u-tubes before reducing system pressure again. The licensee entered the issue into their corrective action program as CR 2008-4131.

  The failure to have an adequate cooldown procedure was more than minor because, if left uncorrected, it could become a more significant safety concern. Specifically, the same procedure would be used during natural circulation operations. Voiding in the steam generator u-tubes under these conditions could challenge the use of the steam generators as a heat sink. Using the NRC Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," and Attachment 1 to Appendix G, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs," the inspectors determined that the finding was of very low risk significance because it did not: 1) result in non-compliance with low-temperature-over-pressure technical specifications; 2) increase the likelihood that a loss of decay heat removal would occur or affect the ability to recover decay heat removal; 3) increase the likelihood of a loss of reactor coolant system inventory or affect the ability to terminate a primary system leak; 4) increase the likelihood of a loss of offsite power or affect the ability to recover from a loss of offsite power; nor 5) affect containment integrity. Also, this finding had a cross-cutting aspect in the area of human performance related to the decision making component because control room personnel failed to use conservative assumptions when deciding to proceed with plant depressurization, considering the unusual circumstance of excessive residual heat in the steam generators and reactor vessel head (H.1(b)) (Section 4OA3).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” for the failure to promptly implement corrective actions for a condition adverse to quality. Specifically, in 1990 the licensee identified that containment spray pumps may runout, and possibly fail, under certain conditions. For example, if one containment spray pump failed for mechanical reasons (such as a shaft failure) the remaining pump would be subjected to runout conditions. Corrective measures were inadequate, in that the potential failure mode continued to exist from 1990 until identified by the inspectors in 2008.

  This finding was greater than minor because it was similar to non-minor example 3.j in NRC Manual Chapter 0612, Appendix E, "Examples of Minor Issues," in that there was a reasonable doubt concerning the operability of the containment spray system, assuming a worst case single failure. Using the NRC Manual Chapter 0609, Phase 1 worksheet, "Initial Screening and Characterization of Findings," the finding screened as having very low safety significance because it did not: 1) represent a degradation of the radiological barrier for the control room, auxiliary building, or spent fuel pool; 2) represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere; 3) represent an actual open pathway in the containment; and 4) involve a degradation of the hydrogen ignitor function. This violation was entered in the licensee’s Corrective Action Program as CR 2008-1683. This finding did not have a crosscutting aspect because the performance deficiency was a long-standing issue and not necessarily indicative of current performance (Section 4OA5.3).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” having very low safety significance for the licensee’s failure to maintain an adequate Boric Acid Corrosion Control Procedure. Specifically, the procedure failed to include requirements identified in their boric acid program basis document, did not provide clear guidance for implementation, and failed to specify systems and components required to be inspected. The licensee has entered this finding into their corrective action program as Condition Report 2008-3014.

  The finding was more than minor because if left uncorrected the finding would become a more significant safety concern due to the corrosive effects of boric acid on carbon steel systems and components. The team identified that the finding screened as very low safety significance (Green) since it did not result in a loss of operability, loss of system safety function, or actual loss of safety function of a single train for greater than its Technical Specification allowed outage time. The finding was also found not to result in an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk-significant per 10 CFR 50.65 for greater than twenty-four hours, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The cause of the finding was related to the crosscutting component of decision making (H.1(a)) associated with roles defined since the manner in which the program was created allowed for confusion, in regards to formally defining authority and roles for decisions affecting nuclear safety and communicating these roles to applicable personnel for implementation of boric acid inspection activities (Section 1R08).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a noncited violation of 10 CFR 20.1902(a) because the licensee failed to post radiation areas in the radwaste building with a conspicuous sign or signs bearing the radiation symbol and the words "Caution, Radiation Area." The licensee posted the radiation area signs only at the entrances to the building instead of at the discrete radiation areas even though the majority of the building was not a radiation area. Dose rates in unposted areas were as high as 14 millirem per hour. Immediate corrective actions included posting the discrete areas as radiation areas. This violation was entered into the corrective action program as Condition Report 2008-2949 and additional corrective actions are still being evaluated by the licensee.

  The failure to post a radiation area is a performance deficiency. The finding was greater than minor because it was associated with the cornerstone attribute (exposure control) and the finding affected the Occupational Radiation Safety cornerstone objective, in that, uninformed workers could unknowingly accrue additional radiation dose. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that the finding was of very low safety significance because it did not involve: (1) as low as is reasonably achievable planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding does not have a crosscutting aspect because of the age of the performance deficiency (Section 2OS1).

- Green. The inspectors identified a noncited violation of Technical Specification 5.8.1.a which resulted from a worker failing to follow procedural requirements. Specifically, on March 18, 2008, the radiation protection count-room technician failed to properly document a personnel contamination event. As immediate corrective action, the licensee completed the skin dose calculation and documented the occurrence in the corrective action program as Condition Report 2008-2904.

  The failure to properly document skin contamination is a performance deficiency. This finding is greater than minor because if left uncorrected the finding would become a more significant safety concern, in that the failure to properly document skin contamination events could result in an individual exceeding the shallow dose exposure limit. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined the finding had very low significance because: (1) it was not an as low as is reasonably achievable finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. Additionally, the finding had a crosscutting aspect in the area of human performance, work practice component [H.4.a], because the workers did not use self- or peer- checking as a human error prevention technique to ensure proper documentation and calculation of skin dose (Section 2OS1).

B. Licensee-Identified Violations

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 actions 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 April 15, 2008, when power was decreased to 50 percent to repair a feedwater pump. On April 17, reactor power was increased to 100 percent. On April 19, the plant was shutdown for a refueling outage. On June 16, the reactor was made critical following completion of the outage. On June 23, reactor power was increased to 100 percent. On June 27, reactor power was decreased to 98 percent to allow for moderator temperature coefficient testing. On June 29, reactor power was increased to 100 percent where the plant remained until the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate Power Systems

a. Inspection Scope

The inspectors verified that plant features, and procedures for operation and continued availability of offsite and alternate AC power systems during adverse weather were appropriate. Specifically, the inspectors: (1) reviewed plant procedures, especially those involving communication and coordination between the site and the transmission system operator, (2) noted the required actions if predicted post-trip voltage would not be acceptable, and (3) reviewed the compensatory actions if the licensee or transmission system operator would not be able to predict post trip voltage.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors completed a review of the licensee's readiness of seasonal susceptibilities involving extreme high winds. The inspectors: (1) reviewed plant procedures, the Updated Safety Analysis Report (USAR), and Technical Specifications (TS) to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the systems listed below to ensure that adverse weather protection features were sufficient to support operability, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee could maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program (CAP) to determine if the licensee identified and corrected problems related to adverse weather conditions.

- May 15, 2008, review of preparations for extreme high winds and possibility of missiles causing a transient or damage to equipment

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Equipment Walk-downs

a. Inspection Scope

Partial Walkdown

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 walk down to the licensee's USAR and CAP to ensure problems were being identified and corrected.

- April 24, 2008, Verification of containment closure

- May 21, 2008, Portions of the Chemical and Volume Control System (CVCS) that include containment penetration M-2

- June 18, 2008, Train 2 of steam generator isolation system circuitry, while main feedwater regulating isolation Valve HCV-1103 was inoperable due to a leak

Documents reviewed by the inspectors included: Operating Instruction OI-CO-4, “Containment Closure,” Revision 45; Omaha Public Power District Drawing E-4220, “Containment Closure Status Board,” Revision 3; Drawing E-23866-210-120, Sheet 1A, “Chemical and Volume Control System P&ID,” Revision 21; and Updated Safety Analysis Report (USAR), Section 7 – Instrumentation and Control.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown (71111.04S)

The inspectors: (1) reviewed plant procedures, drawings, the USAR, TSs, and vendor manuals to determine the correct alignment of the shutdown cooling system; (2) reviewed outstanding design issues, operator workarounds, and USAR documents to determine if open issues affected the functionality of the system; and (3) verified that the licensee was identifying and resolving equipment alignment problems.

Documents reviewed by the inspectors included: Updated Safety Analysis Report (USAR), Sections 6.0 - Engineered Safeguards and Section 9.3 - Shutdown Cooling System; AOP-19, “Loss of Shutdown Cooling,” Revision 13; and OI-SC-1, “Initiation of Shutdown Cooling,” Revision 46.

The inspectors completed one sample.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors walked down the four 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.

- April 30, 2008, Fire Area 30, Containment, Room 1
- April 30, 2008, Fire Area 40, Equipment Hatch Enclosure Area, Room 66
- May 20, 2008, Fire Area 6.3, Basement & Personnel Corridor Area, Room 4
- May 20, 2008, Fire Area 41, Cable Spreading Room, Room 70

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation

a. Inspection Scope

On April 8, 2008, the inspectors observed a fire brigade drill to evaluate the readiness of licensee personnel to prevent and fight fires, including the following aspects: (1) the number of personnel assigned to the fire brigade, (2) use of protective clothing, (3) use of breathing apparatuses, (4) use of fire procedures and declarations of emergency action levels, (5) command of the fire brigade, (6) implementation of pre-fire strategies and briefs, (7) access routes to the fire and the timeliness of the fire brigade response, (8) establishment of communications, (9) effectiveness of radio communications, (10) placement and use of fire hoses, (11) entry into the fire area, (12) use of fire fighting equipment, (13) searches for fire victims and fire propagation, (14) smoke removal, (15) use of prefire plans, (16) adherence to the drill scenario, (17) performance of the postdrill critique, and (18) restoration from the fire drill. The licensee simulated a fire in the seal oil vacuum pump LO-12C,

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Inspection Activities Other Than Steam Generator Tube Inspection, PWR Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The inspection procedure required review of two or three types of nondestructive examination (NDE) activities and, if performed, one to three welds on the reactor coolant system pressure boundary. Inspectors are also guided to review one or two examinations with recordable indications that have been accepted by the licensee for continued service.

The inspectors directly observed the following nondestructive examinations:

System	Component/Weld ID	Exam Type
Reactor Coolant	Reactor Pressure Vessel Closure Head Studs: 8, 9, 10, 11, 12, 13, 14, and 15	UT
Safety Injection	4-CH-13/02, Pipe-to-Elbow Weld	UT
Safety Injection	4-CH-13/04, Elbow-to-Pipe Weld	UT
Safety Injection	2-HPH-2-12/11, Valve-to-Pipe Weld	PT
Safety Injection	2-HPH-2-12/12, Pipe-to-Valve Weld	PT

The inspectors reviewed the following nondestructive examinations through record review:

System	Component/Weld ID	Exam Type
Reactor Coolant	Hot Leg Nozzle-to-Safe End Welds 22 and 28 (RFO-21, Fall 2003)	Ultra-Sonic Testing (UT)
Reactor Coolant	Cold Leg Nozzle-to-Safe End Welds 24, 26, 30, and 32 (RFO-21, Fall 2003)	UT
Reactor Coolant	Reactor Vessel Head to Flange (RFO-24, April 28, 2008)	Visual Test (VT-2)

Additionally, the inspectors reviewed the NDE personnel qualification records for the contractor personnel performing ASME Code Section XI inservice inspections.

The inspection procedure further required verification of one to three welds on Class 1 or Class 2 pressure boundary piping to ensure that the welding process was performed in accordance with the ASME Code. The following Class 2 weld was verified:

System	Component/Weld Identification
Feedwater	BSW-1, Pipe-to-Valve Weld (Class 2)

Welder qualification documentation packages were reviewed for contract welders performing welding activities on the feedwater spool pieces. The documentation packages and logs were in accordance with Article III, QW-300 "Welding Performance Qualification," Section IX of the ASME Code.

The inspectors also verified, by review, that the welding procedure specification (4-MC-GTAW-HT-1, Revision 2) had been properly qualified in accordance with ASME Code Section IX requirements. The inspectors determined that essential variables for the gas tungsten arc welding process (machine) were identified, recorded in the procedure qualification record (PQR 810), and formed the bases for qualification of the welding procedure specification.

Finally, the inspectors verified, by observation and records review, that the welding material had been procured in accordance with ASME Code requirements. Further, since the feedwater piping welds were scheduled to receive a postweld heat treatment, the inspectors verified by review of certified material test reports that the welding material had been appropriately qualified in both the as-welded and postweld heat-treated conditions.

The inspectors completed one sample under Section 02.01.

b. Findings

No findings of significance were identified.

.2 Vessel Upper Head Penetration (VUHP) Inspection Activities

a. Inspection Scope

During Refueling Outage 23, the licensee replaced the original reactor pressure vessel head with a new Alloy 690 head. Since the head had just completed its first operating cycle, the licensee chose to perform a general visual examination of the head and over-head components without removing insulation. Additionally, the licensee conducted a visual examination (VT-2) of the reactor vessel head-to-flange joint. This examination, performed before the inspectors arrived on site, was documented in licensee Inspection Report 2008-0149, dated April 28, 2008, which stated that there was no evidence of leakage. The inspectors reviewed the report and accompanying photographs.

The VT-2 visual examinations were stated to have been performed in accordance with Surveillance Test Procedure QC-ST-MX-3001, “VT-2 Examination of Normally Insulated Class 1 Pressure Retaining Bolted Connections in Systems Borated for Reactivity Control,” Revision 3. Qualifications of the NDE Level II VT-2 Examiner were reviewed and verified to be current.

The inspectors also reviewed Calculation FC 07361, “Calculation of Effective Degradation Years (EDY) for the FORT CALHOUN STATION Reactor Pressure Vessel Head (RPVH) During Cycle 24,” Revision 2, dated April 25, 2008. The calculation, using the methodology specified in NRC Order EA-03-009, showed 1.338, “Effective Full Power Years,” and 0.78, “Effective Degradation Years accrued during Operational Cycle 24.”

The inspectors completed one sample under Section 02.02.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion.

The inspection procedure required review of a sample of boric acid corrosion control walkdown visual examination activities through either direct observation or record review. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control program as specified in the Program Basis Document PBD-10, "Boric Acid Corrosion Prevention," Revision 11. Visual records of the components and equipment were also reviewed by the inspectors.

The inspection procedure required verification that visual inspections emphasize locations where boric acid leaks can cause degradation of safety significant components. The inspectors verified by program/record review that the licensee's boric acid corrosion control inspection efforts were directed towards locations where boric acid leaks can cause degradation of safety-related components. On those components where boric acid was identified, the engineering evaluations were reviewed to ensure the ASME Code wall thickness limits were properly maintained. The evaluations were also reviewed to confirm the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code.

The inspection procedure required both a review of one to three engineering evaluations performed for boric acid leaks found on reactor coolant system piping and components, and one to three corrective actions performed for identified boric acid leaks. The inspectors reviewed 13 engineering evaluations associated with boric acid leaks found since the previous outage. The evaluations consisted of leaks that were identified as major leaks according to the licensee's screening process. The evaluations were reviewed for the causes and corrective actions. The inspectors reviewed 13 condition reports associated with boric acid leaks and confirmed the corrective actions were consistent with established requirements.

The inspectors completed one sample.

b. Findings

Introduction. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," having very low safety significance (Green) for the licensee's failure to maintain an adequate boric acid corrosion control procedure. Specifically, the designated procedure failed to include requirements identified in the licensee's boric acid program basis, did not provide clear guidance for implementation, and failed to specify systems and components required to be inspected.

Description. During the inservice inspection activities, the inspectors reviewed the licensee's boric acid corrosion control program and identified that the methodology for performing the inspection in Procedure SE-EQT-MX-0002, "Carbon Steel and Low Alloy Steel Fasteners Inservice Testing – Refueling Inspections," did not provide detailed guidance for implementation, contained steps that did not include adequate guidance to determine that activities have been satisfactorily accomplished, and referenced additional procedures that contained varying guidance to take credit for performing actions required by the design basis document.

Additionally, the procedure failed to include requirements identified in the program and to provide clear guidance for what will occur once a leak is identified. Procedures that impact activities affecting quality shall include appropriate quantitative and qualitative acceptance criteria. The inspectors noted that the criteria for identifying minor and major leaks was not well defined, was subjective, and defined minor and major differently depending on which procedure was used. Additionally, Procedure SE-EQT-MX-002 addressed boric acid identification but did not address corrective actions with respect to the repetitive identification of the same leaks.

As a result of observations and direct and indirect review, the inspectors questioned whether the principal inspection procedure for the program contained the level of detail necessary for boric acid inspections to be consistently performed. In the licensee's Boric Acid Corrosion Control Program, it was identified that the principal Procedure SE-EQT-MX-002 used to perform inspections did not contain all of the requirements discussed in the design basis document. It was also evident that certain corrective actions have not been adequate, which the inspectors attributed to weaknesses in the implementing procedures.

The following are examples of the identified inadequacies with the boric acid Procedure SE-EQT-MX-002:

- Program Basis Document PBD-10, "Boric Acid Corrosion Prevention," Section 4.4, "Scope," identified nine systems for inclusion in the program. During review of the various implementing inspection procedures, the inspectors noted that there were no components identified from two of the systems (VA-CON: Containment HVAC System and WDL: Waste Disposal Liquid System) although these components were identified in the program basis document.

- Procedure SE-EQT-MX-0002, Attachment 9.1, "Mechanical Closures and Joints with Carbon Steel Fasteners," identified all joints and components to be inspected during performance of this procedure. The inspectors identified that the list did not include numerous susceptible carbon steel and low alloy steel components. Specifically, certain high-pressure safety injection/reactor coolant interface valves were not listed (e.g., HCV-327, -329, -331, and -333).

- Procedure SE-EQT-MX-0002 stated that additional inspection items may be added in Attachment 9.2, "Operating Experience-Additional Inspection Items." The procedure, however, lacked guidance with respect to permanently including components for future inspections. For example, for the inspections performed during March 2005, four components were added to Attachment 9.2 (– LCV-685, AC-216, AC-236, and CH-185). All four were components identified as "major" leakage. However, none were added to Attachment 9.1 from the inspections performed during Fall 2006 and December 2007.

- Procedure SE-EQT-MX-0002, Section 6.5, identified that some components may require inspection per other procedures (e.g., QC-ST-MX-3001, -3002, and SE-ST-SDC-3003) and those inspections should be performed in a manner that would eliminate redundant inspections. The procedure provided a sign off block on this step without clarification as to what work was done and identification of which components were subject to this guidance.

These procedural inadequacies are indicative of the absence of the level of detail necessary for boric acid inspections to be consistently performed by Procedure E-EQT-MX-0002.

Analysis. The inspectors determined that the failure to maintain an adequate boric acid inspection procedure was a performance deficiency and that the finding was more than minor in accordance with Inspection Manual Chapter 0612, Appendix B, "Issue Disposition Screening," because if left uncorrected the finding would become a more significant safety concern. The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as very low safety significance (Green) since it did not result in a loss of operability; loss of system safety function; actual loss of safety function of a single train for greater than its Technical Specification allowed outage time; actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours; and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. Although the procedure inadequacies resulted in the ineffective implementation of the boric acid program, none of the systems and/or components failed to perform their intended safety functions.

The cause of the finding was related to the crosscutting component of Decision Making [H.1(a)] associated with roles defined since the manner in which the program was created allowed for confusion in regards to formally defining authority and roles for decisions affecting nuclear safety and communicating these roles to applicable personnel for implementation of boric acid inspection activities noted in the procedure.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of any type appropriate to the circumstances and shall be accomplished in accordance with these instructions. Contrary to the above, licensee personnel failed to maintain a boric acid corrosion control procedure that adequately addressed activities and thoroughly documented applications appropriate to the circumstances. Because the violation was of very low safety significance and the licensee entered the finding into their CAP, this violation is being treated as an NCV consistent with Section VI.A1 of the NRC Enforcement Policy: NCV 05000285/2008003-01, "Inadequate Boric Acid Control Procedure." The licensee entered the finding into their CAP as Condition Report 2008-3014.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure specified performance of an assessment of in situ screening criteria to assure consistency between assumed nondestructive examination flaw sizing accuracy and data from the Electric Power Research Institute (EPRI) examination technique specification sheets. It further specified assessment of appropriateness of tubes selected for in situ pressure testing, observation of in situ pressure testing, and review of in situ pressure test results. At the time of this inspection, no conditions had been identified that warranted in situ pressure testing.

In addition, the inspectors reviewed both the licensee site-validated and qualified acquisition, analysis technique sheets used during this refueling outage, and the qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy, tubing, equipment, technique, and analysis had been identified and qualified through demonstration. The inspectors-reviewed acquisition technique and analysis technique sheets are identified in the attachment.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess the licensee's prediction capability. Since the replacement steam generators had been installed during the previous refueling outage (RFO-23, Fall 2006), the only credible concern dealt with wear degradation and that was not expected to occur during the first few operating cycles. Stress corrosion cracking is not expected in any areas; primarily because of the tube material composition (i.e., thermally treated Alloy 690). No new damage mechanisms had been identified during this inspection.

The inspection procedure specified confirmation that the steam generator tube eddy current test scope and expansion criteria meet Technical Specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors evaluated the recommended steam generator tube eddy current test scope established by TS requirements and determined that the licensee had, as a minimum, established a test scope that met TS requirements, EPRI guidelines, and commitments made to the NRC. Additionally, the inspection took into account any indications identified during the preservice inspection. The scope of the licensee's eddy current examinations of tubes for all steam generators included:

Steam Generator RC-2A

Bobbin examination – 100 percent of accessible tubes full length (5200 tubes)
Multiple Rotating Pancake Coil examination - special interest inspection program (26 tubes)

Steam Generator RC-2B

Bobbin examination – 100 percent of accessible tubes full length (5199 tubes – one had been plugged during preservice inspection as a preventive measure) Multiple Rotating Pancake Coil examination - special interest inspection program (22 tubes).

The inspection procedures specified, if new degradation mechanisms were identified, verify that the licensee fully enveloped the problem in its analysis of extended conditions, including operating concerns, and had taken appropriate corrective actions before plant startup. To date, the eddy current test results had not identified any new degradation mechanisms.

The inspection procedure requires confirmation that the licensee inspected all areas of potential degradation, especially areas that were known to represent potential eddy current test challenges (e.g., top-of-tubesheet, tube support plates, and U-bends). The inspectors confirmed that all known areas of potential degradation were included in the scope of inspection and were being inspected.

The inspection procedure further requires verification that repair processes being used were approved in the TSs. No repairs (i.e., plugging of tubes) were required.

The inspection procedure also requires confirmation of adherence to the TS plugging limit, unless alternate repair criteria have been approved. The inspection procedure further requires determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspectors determined that the TS plugging limits were being adhered to (i.e., 40 percent maximum through-wall indication).

If steam generator leakage greater than 3 gallons per day was identified during operations or during post shutdown visual inspections of the tubesheet face, the inspection procedure requires verification that the licensee had identified a reasonable cause based on inspection results and that corrective actions were taken or planned to address the cause for the leakage. The inspectors did not conduct any assessment because this condition did not exist.

The inspection procedure requires confirmation that the eddy current test probes and equipment were qualified for the expected types of tube degradation and an assessment of the site-specific qualification of one or more techniques. The inspectors observed portions of eddy current tests performed on the tubes in both Steam Generators. During these examinations, the inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered to, and (4) probe travel speed was in accordance with procedural requirements. The inspectors performed a review of site-specific qualifications of the techniques being used. These are identified in the attachment.

If loose parts or foreign material on the secondary side were identified, the inspection procedure specified confirmation that the licensee had taken or planned appropriate repairs of affected steam generator tubes and that they inspected the secondary side to either remove the accessible foreign objects or perform an evaluation of the potential effects of inaccessible object migration and tube fretting damage. At the time of the inspection, no foreign material had been identified.

Finally, the inspection procedure specified review of one to five samples of eddy current test data if questions arose regarding the adequacy of eddy current test data analyses. The inspectors did not identify any results where eddy current test data analyses adequacy was questionable.

The inspectors completed one sample under Section 02.04.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection scope

The inspection procedure required review of a sample of problems associated with inservice inspections documented by the licensee in the corrective action program for appropriateness of the corrective actions.

The inspectors' review, which dealt with inservice inspection activities, found that the corrective actions were appropriate (see attachment for reviewed condition reports). From this review, the inspectors concluded that the licensee had an appropriate threshold for entering issues into the corrective action program and had procedures that direct a root cause evaluation when necessary. The licensee also had an effective program for applying industry-operating experience.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

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. The training scenario involved a total loss of all feedwater when secondary cooling from auxiliary feedwater was not available.

Documents reviewed by the inspectors are listed in the attachment

The inspectors completed one sample.

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 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 TSs.

- October 2007, Raw water pump packing leak of 40 gpm, specifically the functional failure determination

- June 2008, Reactor Coolant Pumps RC-3A through 3D in a(1) goal monitoring category based on poor performance last cycle

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)

.1 Risk Assessments and Management of Risk

a. Inspection Scope

The inspectors reviewed the four 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.

- April 1, 2008, Review of an orange activity risk color and yellow daily risk color due to diesel-driven auxiliary feedwater pump being out-of-service for an extended period of time

- April 9, 2008, Review of the overall risk assessment plan for the Spring 2008 refueling outage

- April 24, 2008, Review of orange risk condition with the reactor coolant system at midloop

- April 26, 2008, Review of yellow risk while three of four raw water pumps were inoperable while in Mode 5

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four 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 TSs; (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.

- June 30, 2008, Review of the operability of the CVCS system due to potential failures of CH-202

- June 30, 2008, Review of the operability of the containment coolers due to a high energy line break of aux steam piping in Room 69 of the Auxiliary Building

- May 6, 2008, Review of the MSIV closure issue described in CR 2008-2559, where the valve would not fully close

- May 28, 2008, Operability evaluation of containment Air Coolers VA-3 A/B and VA-7A/B and the associated Dampers VA-56A/B, VA57-A/B under backdraft conditions

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the following temporary/permanent modifications to verify that the safety functions of important safety systems were not degraded:

- May 3, 2008, Installation of a jumper on the polar crane to allow the trolley to move to the hard stops. Modification was removed on May 4, 2008

- June 18, 2008, Permanent plant modification involving feedwater regulating controls replacement

- May 18, 2008, Implementation of the permanent plant modification uprating the shutdown cooling system entry conditions

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples during the inspection.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the five post-maintenance 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 post-maintenance testing.

- April 8, 2008, Postmaintenance test of FW-54, Diesel Auxiliary Feedwater Pump, following replacement of a cylinder

- May 27, 2008: Postmaintenance test of pressurizer pressure low signal actuation and blocking logic

- May 29, 2008, Postmaintenance testing of control rods including position indication checks

- June 4, 2008, Postmaintenance testing following replacement of HCV-1105, Steam Generator RC-2A feedwater regulating bypass valve

- June 19, 2008, Postmaintenance testing on raw water Pump AC-10D following the intake cell being out of service

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 TSs, 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) heat-up and cool-down activities; (13) restart activities; and (14) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities. The inspectors' containment inspections included observations of the containment sump for damage and debris; and supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the USAR, procedure requirements, and TSs to ensure that the six surveillance activities listed below demonstrated that the SSC's 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 TS 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 SSC's not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- April 16, 2008, Observation of the set pressure test of main steam safety Valve MS-279 (Inservice Testing)

- May 27 through June 3, 2008, observation of testing and the initial failure of containment spray and low pressure safety injection pumps and an in-office review of the reperformed test

- May 28, 2008 Observed completion of Procedure OP-ST-SI-3007, "High Pressure Safety Injection system Pump and Check Valve Test," Revision 24

- May 29, 2008, Review of the integrated leak rate testing performed at the end of 2008 refueling outage (routine containment isolation valve testing)

- June 2, 2008, Observation of the functional testing of 4160 volt breakers for auto start prohibit and undervoltage trip prohibit

- June 4, 2008, In-office review of Procedure OP-ST-SI-3007, "High Pressure Safety Injection System Pump and Check Valve Test," Revision 24, where check Valve SI-344 failed during the surveillance test

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six 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 an in-office review of "Emergency Facilities and Equipment," Revision 35 to Section H, to the Fort Calhoun Station Radiological Emergency Response Plan, submitted March 3, 2008. This revision moved the licensee's Operations Support Center inside the ventilation and radiation-shielding envelope of the Technical Support Center.

The revision was compared to the previous revision, 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, and to the standards in 10 CFR 50.47(b) to determine if the revision adequately implemented the requirements of 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of changes made by the licensee; 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, and worker adherence to these controls. The inspectors used the requirements in 10 CFR Part 20, the TSs, and the licensee's procedures required by TSs 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, or airborne radioactivity areas

- 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 airborne radioactivity areas

- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools

- 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

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination control 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. A Green NCV of 10 CFR 20.1902(a) was identified for the failure to conspicuously post discrete radiation areas in the radwaste building with a sign or signs bearing the radiation symbol and the words "Caution, Radiation Area."

Description. On May 1, 2008, while touring the radwaste building, the inspectors identified three discrete localized radiation areas, which were not posted as radiation areas. The general area dose rates were as high as 14 millirem per hour. The only radiation area signs to warn workers prior to entering the radiation areas were at the entrances to the radwaste building. However, the radwaste building is a large area, and according to the licensee's surveys, only a small part of the total area had dose rates exceeding 5 millirem per hour. Immediate corrective actions included posting the discrete areas as radiation areas. Additional corrective actions are still being evaluated by the licensee.

Analysis. The inspectors reviewed the applicable guidance in NUREG/CR-5569, Revision 1, Health Physics Positions 036, "Posting of Entrances to a Large Room or Building as a Radiation Area," and 066, "Guidance for Posting Radiation Areas." Since the area was large, and only isolated areas were actually radiation areas, the inspectors concluded that posting only the entrances to the area, rather than the discrete areas, was not sufficient to inform radiation workers of the radiological hazards in the work areas. The failure to post a radiation area is a performance deficiency. The finding was greater than minor because it was associated with the occupational radiation safety exposure control attribute and affected the cornerstone objective, in that, uninformed workers could unknowingly accrue additional radiation dose. Since the finding involved the potential for unplanned, unintended dose resulting from conditions that were contrary to NRC regulations, the finding was evaluated using the "Occupational Radiation Safety Significance Determination Process." The inspectors determined that the finding was of very low safety significance because it did not involve: (1) as low as is reasonably achievable planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding does not have a crosscutting aspect because of the age of the performance deficiency.

Enforcement. A radiation area is defined, in 10 CFR 20.1003, as an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 millirem in an hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates. As stated in 10 CFR 20.1902(a), the licensee shall post each radiation area with a conspicuous sign or signs bearing the radiation symbol and the words "Caution, Radiation Area." Contrary to these requirements, on May 1, 2008, three discrete radiation areas in the radwaste building were not conspicuously posted as radiation areas. As a corrective action, the licensee immediately posted each of the three discrete areas. Additional corrective action is being evaluated. Due to the failure to conspicuously post the radiation areas was determined to be of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2008-2949, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000285/2008003-02, "Failure to Conspicuously Post a Radiation Area."

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:

- Site-specific ALARA procedures

- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements

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

- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered

- 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

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

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

- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

The inspectors completed 11 of the required 29 samples.

b. Findings

Introduction. During a review of the Personnel Contamination Log on April 30, 2008, the inspectors identified a NCV of TS 5.8.1.a for failure to follow a licensee procedure.

Description. On March 18, 2008, an individual who had been assisting with replacement of a bladder on Valve CH-29B was found to have contamination on the right side of their head. The contamination was determined to have an activity of 5000 net counts per minute (NCPM). The particle was removed from the individual's head and an isotopic analysis performed to determine the nuclide and activity level. The particle was found to be 0.241 microcuries of Cobalt-60. The event was not documented on a Personnel Contamination Report as required by station procedures. Immediate corrective actions included calculation of skin dose using the results of the isotopic analysis performed at the time of the contamination, and revision of the associated procedure to more clearly define the required action levels for documentation of skin contamination events.

Analysis. The failure to properly document skin contamination is a performance deficiency. This finding is greater than minor because, if left uncorrected the finding would become a more significant safety concern, in that, the failure to properly document skin contamination events could result in an individual exceeding the shallow dose exposure limit. Utilizing Manual Chapter 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” the inspectors determined that the finding was of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Additionally, the finding had a crosscutting aspect in the area of human performance, work practice component, [H.4.a] because the workers did not use self- or peer-checking as a human error prevention technique to ensure proper documentation and calculation of skin dose.

Enforcement. Technical Specification 5.8.1.a requires procedures to be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Appendix A. Appendix A Section 7 recommends radiation protection procedures for personnel monitoring. Licensee Procedure RPI-1, “Personnel Monitoring and Decontamination,” Section 7.5.2.C.1 states, in part, “All skin and/or clothing contamination events at Action Level II or greater (greater than or equal to 5000 NCPM) shall be documented on FC-RP-207-1, Personnel Contamination Report.” Contrary to this requirement, on March 18, 2008, the count-room technician failed to initiate a Personnel Contamination Report and calculate the resulting skin dose. Due to the failure to follow a procedure is of very low safety significance and has been entered into the licensee’s corrective action program as Condition Report 2008-2904, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000285/2008003-03, “Failure to Follow Procedures.”

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Cornerstone: Barrier Integrity

a. Inspection Scope

The inspectors sampled submittals for the performance indicators listed below for the period covering October 1, 2007, through April 18, 2008. The definitions and guidance of Nuclear Engineering Institute 99-02, “Regulatory Assessment Indicator Guideline,” Revisions 2 through 4, were used to verify the licensee’s basis for reporting each data element in order to verify the accuracy of performance indicator data reported during the assessment period.

- Reactor coolant system specific activity
- Reactor coolant system leakage

b. Findings

No findings of significance were identified.

.2 Cornerstone: Occupational Radiation Safety

Occupational Exposure Control Effectiveness

The inspector reviewed licensee documents from October 1, 2007, through March 31, 2008. 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 Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 5). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

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

Cornerstone: Public Radiation Safety

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

The inspector reviewed licensee documents from October 1, 2007, through March 31, 2008. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

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

Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Reviews of Identification and Resolution of Problems

a. Inspection Scope

In March 2008, a team performed Inspection Procedure 95002, "Inspection for One Degraded Cornerstone or any Three White Inputs in a Strategic Performance Area." The purpose of the supplemental inspection was to examine the problem identification, root cause evaluation, extent of condition and extent of cause determination, and corrective actions associated with multiple White issues.

The team determined that Fort Calhoun Station had failed to adequately address the White finding associated with inadequate maintenance procedures and post-maintenance testing of auxiliary contacts on an emergency diesel generator. Specifically, the assessment of the failed auxiliary contacts did not adequately address a potential generic failure mechanism of a sticking contact actuator. This unaddressed failure mechanism could be caused by the inappropriate application of wet lubricant and build up of dust and debris. Past maintenance practices resulted in the application of wet lubricant that was a significant contributor to the failure of the emergency diesel generator. Additionally, at the time of the inspection, the timetable of actions to address the scope of extent of condition to other relays and contacts was not considered timely given the potential common mode failure mechanism (high resistance contacts due to poor past maintenance practices). Additionally, the licensee was also actively engaged with the development and refinement of preventative maintenance strategies for relays and contactors at the time of the inspection.

Consequently, the NRC was not able to effectively evaluate the robustness and adequacy of the licensee's preventative maintenance plans at the time of the inspection. As a result, the White finding associated with Notice of Violation 05000285/2007011-03, "Failure to Provide Procedure for Safety-Related Maintenance Activities," remained open. The inspectors, during this inspection, verified that: (1) the concerns of extent of condition of inadequately maintained relays and contacts were appropriately assessed; and (2) the action items relative to future preventative maintenance of risk important components and subcomponents (such as electrical relays and contactors) were adequate. When evaluating the effectiveness of the licensee's corrective actions, the following attributes were verified:

- Corrective actions to visually inspect auxiliary contacts, including looking for sticking contacts, were planned and were being performed by the licensee

- Objective criteria were established in determining when auxiliary contacts were in a failed condition

- Timeliness of the corrective actions planned were commensurate with the safety-significance of the components

- Depth/breadth of the corrective actions was appropriate (i.e., extent of condition and extent of cause)

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Review of Identification and Resolution of Aspects of Reactor Coolant System Voiding Issues

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's problem identification and resolution process with respect to the reactor coolant system steam-voiding event on June 10, 2008. (Please refer to Section 4OA3 of this inspection report for a description of this event.)

The inspectors reviewed the following Condition Reports: 199600340 dated May 21, 1996; 199800656 dated April 5, 1998; 199800660 dated April 5, 1998; 199800739 dated April 11, 1998; and 20084131 dated June 12, 2008

The inspectors completed one sample during the inspection.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 Steam Voiding in the Reactor Coolant System During Plant Cooldown

a. Inspection Scope

The inspectors reviewed the circumstances surrounding and licensee response to a steam void formed during plant cooldown on June 10, 2008. The inspectors reviewed the licensee's condition reports, logs, and graphs of key plant parameters, related operating experience, and associated procedures. In addition, the inspectors interviewed personnel from the operating crew on shift during the event and reviewed their written testimonies to assess operator actions against procedural requirements and to assess the adequacy of the plant procedures.

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

The inspectors completed one sample during the inspection.

b. Findings

Introduction. A Green self-revealing NCV of TS 5.8.1.a was identified for the failure to have an adequate procedure for plant cooldown. Specifically, on June 10, 2008, the plant cooldown procedure allowed the control room staff to unexpectedly draw an approximately 2700-gallon steam void in the reactor coolant system. The procedure failed to provide guidance to ensure the reactor vessel head and steam generator u-tubes were sufficiently cooled down before depressurizing the reactor coolant system. Contributors to the event included: (1) the failure to institutionalize related operating experience from NRC Generic Letter 81-21, "Natural Circulation Cooldown," dated May 5, 1981; and (2) the failure of plant operators to implement related training intended to avoid void formation. After voids formed, operators recognized the void indications, raised system pressure to collapse the steam voids, and then cooled the vessel head and steam generator u-tubes before reducing system pressure again.

Description. On June 10, 2008, control room operators implementing the plant shutdown Procedure OP-3A and the pressurizer cooldown and venting Procedure OI-RC-4A accidentally drew an approximate 2700-gallon steam void in the reactor coolant system (RCS). The plant had been heating up for post-outage startup when a packing leak was discovered on the Pressurizer Spray Valve PCV-103-2 leak-off line. In order to affect repairs on the packing leak, the plant was cooled back down. However, rather than cool all the way down to normal cold shutdown temperatures (80^{o}F-90^{o}F), the licensee elected to only cool the plant down to a temperature band of 180^{o}F-190^{o}F to allow a more rapid return to startup conditions after the packing leak repair. Although this was an abnormal plant state, both Procedures OP-3A and OI-RC-4A allowed the operators to limit the plant cooldown in this manner. Neither the procedures nor the operators anticipated the potential for RCS voiding that limiting the cooldown in this manner created. Once shutdown cooling (SDC) system entry conditions were achieved, the reactor coolant pumps (RCPs) were secured and the steam generators (SGs) were isolated. At that time, the RCS was at approximately 250 psia and approximately 280^{o}F and the SGs were at approximately 35 psia and approximately 260^{o}F.

The SDC system circulates cooling water through the reactor vessel core region and hot leg piping, but leaves the RCS water in the SG u-tubes and reactor vessel head region stagnant. Since neither Procedures OP-3A nor OI-RC-4A required the RCPs to continue to force cooling water into the reactor vessel head to lower its temperature below saturation, enough residual heat was left in the reactor vessel head region to result in some boiling in the head region when the RCS was subsequently depressurized. Similarly, neither procedure required the SGs to be steamed down to cool the RCS water in the u-tubes below saturation for a depressurized RCS, so the heat in the SGs raised the temperature of the RCS in the u-tubes enough to result in boiling in the RCS side of the u-tubes when the RCS was subsequently depressurized. Therefore, at the beginning of the pressurizer cooldown and venting, the RCS was in a condition where SDC was maintaining the core region and hot legs of the RCS subcooled. However the stagnant RCS water in the reactor vessel head and in the SG u-tubes was hot enough that boiling would occur once the RCS pressure was reduced to approximately 24 psia during the pressurizer venting steps. The procedures neither prevented this set of conditions nor provided sufficient notes and cautions to ensure that the crew would recognize the likelihood of boiling in the RCS. The crew was trained that voiding in the reactor vessel head region would be indicated by an alarm on the reactor vessel level management system (RVLMS) but was not trained to use the RVLMS thermocouples as a direct indication of temperatures in the reactor vessel head region that would indicate the onset of saturation conditions. Failing to utilize the thermocouples as direct indications of saturation meant that a void of approximately 688 gallons would have to form in the head region before the RVLMS system would alarm and warn the crew of void formation.

As a result of these plant conditions, several times during the subsequent pressurizer, cooldown procedure RCS pressure was lowered to the point that boiling in the upper vessel head region and SG u-tubes occurred. Then, at the completion of the pressurizer cooldown, when the pressurizer was vented to the pressurizer quench tank and operators began draining coolant from the RCS to backfill the pressurizer with nitrogen, per Procedure OI-RC-4, an approximate 2700 gallon steam void was formed in the SG u-tubes (which was, at this time, the portion of the RCS closest to saturation conditions and therefore the first to begin voiding as pressure was reduced) before operators recognized the indications of RCS steam void formation and secured the drain down. In fact, shortly after establishing the desired RCS draindown flow rate, the operators began to question whether quench tank pressure (which should have decreased but didn't) and pressurizer level (which should have come into the indicating range after approximately 15 minutes of draining but didn't) were tracking as expected. However, the operators continued the draindown for approximately 32 minutes and even increased the draindown flow rate 15 minutes into the evolution before concluding that a void must be forming in the RCS. During the void formation, RCS pressure was approximately 24.7 psia while RCS temperature in the SG u-tubes was approximately 260^{o}F (calculated saturation temperature for that pressure was 243^{o}F) and RCS temperature in the upper reactor vessel head region was approximately 263^{o}F (calculated saturation temperature was approximately 261^{o}F – note that the head region had an additional 25 feet of static fluid head because of its lower elevation in the RCS and this results in a higher saturation temperature than the SG u-tube region). Since the temperature in the head region was very close to the calculated saturation temperature, it is unclear how much boiling actually occurred in the head region, but any boiling that did occur there did not form a void larger than approximately 688 gallons since a larger void would have generated a RVLMS alarm. Clearly, the majority of boiling and void formation occurred in the SG u-tube region. The size of the resultant void was approximately 2700 gallons, was calculated based on the rate that fluid was drained from the RCS and the time that the draining occurred (approximately 65 gpm for 15 minutes and approximately 100 gpm for 17 additional minutes). However, void formation in stagnant legs of the reactor coolant system at Fort Calhoun Station does not create a significant threat to shutdown cooling system operation since the void would have had to completely empty the RCS below the hot leg in order to threaten the flow path or the net positive suction head of the SDC pumps.

The licensee increased RCS pressure to collapse the steam voids and cooled down the reactor vessel head region and the SGs before resuming pressurizer draindown and venting operations.

Analysis. The failure to have an adequate shutdown procedure is a performance deficiency. The finding was more than minor because, if left uncorrected, it could become a more significant safety concern. Specifically, the same procedure would be used during natural circulation operations. Voiding in the steam generator u-tubes under these conditions could challenge the use of the steam generators as a heat sink. Using the NRC Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," and Attachment 1 to Appendix G, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs," the inspectors determined that the finding was of very low risk significance because it did not: (1) result in noncompliance with low-temperature-over-pressure technical specifications; (2) increase the likelihood that a loss of decay heat removal would occur or affect the ability to recover decay heat removal; (3) increase the likelihood of a loss of reactor coolant system inventory or affect the ability to terminate a primary system leak; (4) increase the likelihood of a loss of offsite power or affect the ability to recover from a loss of offsite power; nor (5) affect containment integrity. Also, this finding had a crosscutting aspect in the area of human performance related to the Decision Making Component because control room personnel failed to use conservative assumptions when deciding to proceed with plant depressurization, considering the unusual circumstance of excessive residual heat in the steam generators and reactor vessel head [H.1(b)].

Enforcement. Technical Specification 5.8.1.a requires written procedures to be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Revision 2, Appendix A.2.j, recommends operating procedures for plant shutdown from hot standby to cold shutdown be written. Contrary to this requirement, on June 10, 2008, the voiding event revealed that Omaha Public Power District's Fort Calhoun Station failed to establish an adequate written operating procedure for plant shutdown. Specifically, Procedure OP-3A, "Plant Shutdown," Revision 73, did not contain adequate guidance to prevent the formation of steam voids in the RCS reactor vessel head and SG u-tubes. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CR 20084131, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000285/2008003-04, "Failure to Write an Adequate Shutdown Procedure."

.2 (Closed) LER 05000285/2008001-00, Reactor Trip Due to Turbine Control System Failure

On March 15, 2008, a circuit board in the electro-hydraulic control system of the main turbine failed. This failure caused Turbine Control Valves CV-1 and CV-3 to shut and resulted in a reactor trip due to the loss of load. The failed turbine control system component was replaced. Postmaintenance testing was performed to ensure reliable operation of the system and the plant returned to full power. The LER was reviewed by the inspectors, no findings of significance were identified, and no violation of NRC requirements occurred. The licensee documented the failed equipment in Condition Report 2008-1592. This LER is closed.

.3 (Closed) LER 05000285/2008002-00, Loss of Containment Integrity Due to a Leaking Isolation Valve

On March 15, 2008, at 08:33 a.m., following a reactor trip from 85 percent power, relief Valve CH-223 lifted and failed to close causing a 2-gallon per minute reactor coolant system leak through the letdown system to the pressurizer quench tank. Valve CH-223 is located on a branch line between two automatic containment isolation valves and is therefore part of the containment boundary. The operators did not immediately recognize Valve CH-223 as a containment boundary valve governed by Technical Specifications. On March 16, 2008, at 1:55 p.m., operators determined Valve CH-223 to be a containment boundary valve and shut the component at 2:01 p.m. The licensee determined the cause to be a failure to translate containment integrity design requirements from the Final and Updated Safety Analysis Reports into appropriate operating procedures and guidance. This finding is more than minor because it affected the Procedure Quality attribute of the Barrier Integrity Cornerstone. This finding was considered to have very low safety significance (Green) because of the small size of the opening (2-inch relief valve) which was connected to a closed system. Therefore, the finding screened as

Green using Table 4.1 of Appendix H to Inspection Manual Chapter 0609. This licensee-identified finding involved a violation of TS 2.6(1)a, “Containment System.” The enforcement aspects of the violation are discussed in Section 4OA7 of this report. This LER is closed.

.4 Review of Reportability of Loss of Shutdown Cooling Event and Subsequent Retraction

On May 20, 2008, at 7:56 p.m. during reactor core reload with the reactor cavity full, electrical power was lost to a nonvital instrument bus. This deenergized bus resulted in a loss of power to shutdown cooling temperature control Valve HCV-341. This event was initially reported by the licensee as a loss of shutdown cooling per Event Notification 44228. The licensee subsequently retracted the report based on (1) the availability of the shutdown cooling system not having been lost during the event and (2) plant procedures having provisions to control the system locally. The inspectors verified that the Control Valve HCV-341 was in fact shut prior to the event meaning that cooling had not been interrupted. Further, through interviews with the operators on watch at the time, the inspectors verified the short time taken to establish positive manual control of the valve and that plant procedures were effective in directing the required actions. No findings of significance were identified and no violation of NRC requirements occurred.

4OA5 Other Activities

.1 Quarterly Resident Inspectors Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors performed observations of security force personnel and activities to ensure that the activities were consistent with Fort Calhoun Station security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspector’s normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item 05000285/2008006-01, High Contact Resistance on Main Steam Bypass Valve Relay Contactors

a. Inspection Scope

During the 95002 inspection, the team reviewed the licensee’s extent of condition of high contact resistance to other components (i.e., those not related to the emergency diesel generators). During the licensee's forced outage in March 2008, inspections identified four components with as-found contact resistance that exceeded the licensee's established acceptance criteria of less than 1 ohm. These components included main steam bypass Valves HCV-1041C and HCV-1042C, volume control tank outlet Valve LCV-218-2, and the high head safety injection to chemical volume control system crosstie isolation Valve HCV 308. The auxiliary contacts for Valve HCV-308 were replaced and the valve declared operable. No immediate safety concerns existed for the other components. The licensee determined that two of those relays (FID-2 components) associated with main steam bypass Valves HCV-1041C and HCV-1042C needed further assessment to demonstrate operability. As an interim action, the licensee tagged the valves in their closed safety position. At the time of the inspection, the licensee's final assessment was pending until the shutdown of the facility for the 2008 refueling outage to allow for as-found testing of the valves. Consequently, an unresolved item (URI) was opened to review any potential regulatory and risk implications (URI 05000285/2008006-01, "High Contact Resistance on Main Steam Bypass Valve Relay Contactors"). The inspectors reviewed the as-found condition of Valves HCV-1041C and HCV-1042C and verified that they had been operable and able to perform their safety function if called upon.

b. Findings

No findings of significance were identified.

.3 (Closed) URI 05000285/2008006-02, Containment Cooling Design Requirements and Licensing Review

a. Inspection Scope

During the 95002 inspection, the team reviewed LER 05000285/2007004-00, "Inadvertent Isolation of Containment Spray due to Inadequate Test Procedure." As a part of that review, the inspectors identified concerns with the licensee's ability to withstand a single failure of the containment spray system. URI 05000285/2008006-02, "Containment Cooling Design Requirements and Licensing Review," was issued to follow-up on this concern.

b. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to promptly implement corrective actions for a condition adverse to quality. Specifically, in 1990 the licensee identified that containment spray pumps may runout, and possibly fail, under certain conditions. For example, if one containment spray pump failed for mechanical reasons (such as a shaft failure) the remaining pump would still be subjected to runout conditions. Corrective measures were inadequate, in that the potential failure mode continued to exist from 1990 until identified by the inspectors in 2008.

Description. On March 17, 2008, during a review of LER 2007-004, "Inadvertent Isolation of Containment Spray due to Inadequate Test Procedure," the team performing Inspection Procedure 95002 postulated that a single mechanical failure could result in the inability of one pump to provide any containment spray flow. For example, a containment spray pump coupling failure, a pump discharge check valve failure to open, or a pump 480 VAC breaker mechanical failure could result in the inability of one pump to provide any containment spray flow. Consequently, the remaining operating pump would operate with both containment spray header isolation Valves HCV-344 and HCV-345 open resulting in a pump runout condition due to the single active mechanical failure.

The licensee previously identified in 1990 that an event in which both Valves HCV-344 and HCV-345 were open simultaneously with only one pump running would cause a runout condition and cause the motor to draw more amperage than the vendor allowable criteria (greater than 15 percent above its nameplate rating). After identifying this issue in 1990, the licensee planned to implement a piping modification to the spray header to prevent the runout condition, but elected instead to install additional valve opening logic. This corrective action was intended to only allow one pump to operate with one spray header valve open or both pumps and both spray headers open. This modification was first installed between the containment spray Pumps SI-3B and SI-3C 480 VAC breakers and the Train A spray header isolation Valve HCV-344. This modification was also installed in 2006 between containment spray Pump SI-3A, and the spray header isolation Valve HCV-345, as well as, removing the auto start feature from the containment spray Pump C. After reviewing Condition Reports 200601606, 200701647, 200701647, and LER 2007-004, the inspectors determined that the licensee's corrective actions were inadequate. Specifically, the inspectors found that the licensee's engineering reviews were focused on electrical aspects of single active failures resulting in pump runout and did not consider single active mechanical failure modes. The inspectors noted that a previously submitted License Amendment 235, which removed the automatic start feature to the containment spray Pump C, also did not consider single active mechanical failures and the potential for pump runout.

The licensee subsequently developed an operability evaluation that credited existing operator actions in the EOPs to secure one of the two running containment spray pumps early in an accident, if containment cooling heat removal requirements were met, as well as providing that operators had been previously trained to identify and take actions to prevent containment spray pump runout. URI 05000285/2008006-02, "Containment Cooling Design Requirements and Licensing Review" was opened during the 95002 Team Inspection regarding the licensing and design requirements of the containment cooling design, including the containment spray system and containment coolers, and the applicability and accuracy of License Amendment 235 to this issue.

Analysis. The failure to promptly correct a condition adverse to quality was a performance deficiency. This finding was greater than minor because it was similar to nonminor Example 3.j in NRC Manual Chapter 0612, Appendix E, "Examples of Minor Issues," in that there was a reasonable doubt concerning the operability of the containment spray system, assuming a worst case single failure. Using the NRC Manual Chapter 0609, Phase 1 Worksheet, "Initial Screening and Characterization of Findings," the finding screened as having very low safety significance because it did not: (1) represent a degradation of the radiological barrier for the control room, auxiliary building, or spent fuel pool; (2) represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere; (3) represent an actual open pathway in the containment; and (4) involve a degradation of the hydrogen ignitor function. This violation was entered in the licensee's corrective action program as CR 2008-1683. This finding did not have a crosscutting aspect because the performance deficiency was a long-standing issue and not necessarily indicative of current performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance's are promptly identified and corrected." Contrary to this requirement, the licensee failed to implement adequate corrective action to correct a design deficiency, which had the potential to render both trains of containment spray unavailable in an event. Because this violation was of very low safety significance and it was entered in the licensee's corrective action program as CR 2008-1683, it is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/2008003-05, "Inadequate Corrective Actions for a Containment Spray Design Deficiency." URI 05000285/2008006-02, "Containment Cooling Design Requirements and Licensing Review," is being closed to this violation.

.4 (Closed) Temporary Instruction 2515/166, "Pressurized Water Reactor Containment Sump Blockage", Fort Calhoun Station, Unit 1

The licensee requested an extension from the NRC for completion of actions concerning Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," as a result of analyses, testing, and design evaluations not being fully complete. The licensee will provide an update to the NRC within 90 days following the Spring 2008 refueling outage.

Listed below are all of the commitments made by the licensee and the manner in which they were disposition (individually and combined). Inspectors verified implementation of committed plant modifications and procedure changes had received prior approval, and that all changes requiring NRC approval had been submitted. On March 4, 2005, the licensee formally responded within 90 days to Generic Letter 2004-02. In this response, the licensee identified the following commitments:

a. An analyses will be performed to determine the susceptibility of the emergency core cooling system and containment spray system recirculation functions to the adverse effects of post accident debris blockage and operation with debris-laden fluids by September 1, 2005, except for the analysis of the debris head loss across the strainers

b. A preliminary debris head loss analysis will be completed by September 1, 2005

c. Details of the methodology used by the licensee will be provided in the Generic Letter 2004-02 response that is due on September 1, 2005

d. The final debris head loss analysis will be completed as part of the strainer modification process by December 31, 2007

All of these commitments have been met.

On June 9, 2005, the licensee submitted a revised request for an extension to the completion date for the corrective actions taken in response to Generic Letter 2004-02. Outstanding (in addition to commitments) is the full resolution of issues associated with chemical and downstream effects that may affect the ultimate size of the replacement strainers. The commitments in this letter involved completing the following corrective measures during the 2006 refueling outage:

a. Replacement of trisodium phosphate with an alternate pH buffer, which reduces the risk for sump screen blockage caused by formation of chemical precipitates (this is being accomplished through a separate license amendment request)

b. Installation of two interim strainer modules (one per train) with approximately 1100 ft^2 of total surface area

c. Removal of the automatic start feature for one containment spray pump (This is being accomplished in a separate license amendment request)

d. Installation of debris exclusion devices on reactor cavity and refueling cavity drain lines

e. Installation of reactor vessel spacer rings to reduce the water hold-up in the upper cavity

f. Replacement of the existing steam generators, pressurizer and reactor vessel head, resulting in replacement of approximately 823 ft^3 of calcium silicate insulation, and removal of approximately 7041 ft^2 of unqualified coatings. This represents removal of approximately 62 percent of the calcium silicate insulation behind the biological shield that may fall within the zones of influence and approximately 35 percent of the unqualified coatings

g. Replacement of calcium silicate insulation on the pressurizer spray line to eliminate generation of calcium silicate debris from the small break loss of coolant accident that presents the greatest risk of debris generation and transport

An additional commitment involved the fact that the latent debris collection procedure will be fully implemented prior to the completion of the 2006 refueling outage. At the time of this inspection, all of these commitments had been met.

As of August 31, 2005, commitments to be completed by December 31, 2007, included:

a. The Fort Calhoun Station emergency core cooling system and containment spray system recirculation functions will be in compliance with the regulatory requirements listed in the applicable regulatory requirements section of the subject generic letter under debris loading conditions

b. A Generic Letter 2004-002 closeout response will be submitted and the final debris loaded head loss margin provided. This will include chemical effects

The November 18, 2005, response duplicated all of the commitments including the plan to fully implement the latent debris collection procedure made on June 9, 2005, with the exception of a change to the values noted in the sixth commitment labeled as f on the list:

> Replacement of the existing steam generators, pressurizer and reactor vessel head, resulting in replacement of approximately 760 ft^3 of calcium silicate insulation with reflective metal insulation and removal of approximately 7100 ft^2 of unqualified coatings (both values are based on preliminary debris calculations).

The following list consists of specific compensatory actions that are associated with the containment sump modification:

a. Enhancement of procedures associated with refilling the safety injection refueling water tank to provide a hierarchy of flow paths depending on equipment availability

b. Establishment of procedural guidance for throttling high pressure safety injection flow after the recirculation actuation signal, to a value that is acceptable to the safety analysis, but less than full flow

c. Enhancement of procedures to identify equipment and instrumentation that could be affected by flooding the containment above the current flood level assumed for equipment qualification

d. Enhancement of procedures to measure water level in containment above the maximum water level at the start of recirculation

e. Training on these enhancements and retraining on the existing compensatory measures

At the time of this inspection, the licensee had received NRC approval (safety evaluation report) and proceeded with their modifications for the containment spray system actuation logic, which would allow for crediting the safety-related fan coolers and eliminating automatic containment spray initiation for a loss-of-coolant accident.

In the latest supplemental response dated February 29, 2008, the remaining six actions are to be completed within 90 days of completion of the Spring 2008 refueling outage except for Items a and b which are expected to be completed prior to startup from the Spring 2008 refueling outage:

a. Confirm if existing cyclone separators are acceptable or replace as needed. Status: The licensee will be replacing them since they did not meet test acceptance criteria.

b. Enhance Standing Order SO-25, "Temporary Modification Control," regarding configuration control of insulation in containment. Status: The procedure change was submitted May 1, 2008.

c. Evaluate the final conditions issued by the NRC in regards to Westinghouse Topical Report WCAP-16793-NP and provide a formal response. Status: At the time of the inspection, the licensee was awaiting NRC approval.

d. Validate flashing evaluation utilizing NRC Safety Evaluation for Licensee Amendment Request LAR-07-04. Status: At the time of the inspection, the licensee was awaiting NRC approval.

e. Validate strainer head loss test results and obtain final report from vendor. Status: At the time of the inspection, the licensee had received and was in the process of review.

f. Provide Generic Letter 2004-02 closeout letter. Status: At the time of the inspection, the closeout letter was incomplete and awaiting completion of all modifications.

Additional modifications, not specified in the licensee's Generic Letter 2004-02 responses are discussed in Modification Engineering Change EC 40070 documentation. As noted above, based on the new configuration, the containment spray system will no longer operate for the mitigation of a loss-of-coolant-accident event. Instead, the air coolers will be credited for the containment pressure and temperature mitigation function following a loss-of-coolant-accident. Specifically, changes/modifications will be implemented for the following items:

a. Containment spray system interlock

b. SGLS logic modifications

c. HCV-480/484 and HCV-481/485 RAS interlock modification

d. Containment Spray Isolation Valves HCV-344 And HCV-345 "Valve Not Closed" Alarm

e. HCV-400A/C, 401A/C, 402A/C and 403A/C low flow interlock modification

The inspection phase of Temporary Instruction 2515/166 for Fort Calhoun Station is complete.

.5 <u>Temporary Instructions 2515/172, "Reactor Coolant System Dissimilar Metal Butt Welds" Fort Calhoun Station, Unit 1</u>

During the Fall 2006, Refueling Outage 23, the licensee replaced steam generators, the pressurizer, and the reactor pressure vessel head. All of the associated Alloy 600 nozzles/penetration locations and Alloy 82/182 butt welds (the subject of Materials Reliability Program (MRP) 139) have been eliminated with the installation of the new equipment. The new welded joints/materials will be examined in accordance with the requirements of ASME Code, Section XI, IWB-2500 requirements.

The available information (i.e., original construction drawings) was indeterminate with respect to the welding materials used in the reactor pressure vessel hot leg and cold leg nozzles. During the current Spring 2008, Refueling Outage 24, the licensee contracted to have a definitive eddy current examination performed on the two hot leg nozzles, which would establish whether Alloy 82/182 butt welds existed. The eddy current examinations are able to distinguish permeability changes between carbon steel, stainless steel, and Alloy 82/182 materials, as evidenced by use of a test coupon fabricated specifically for that purpose. The eddy current examinations used three different probes: low frequency, rotating pancake, and plus point. The plus point probe was not adequate for identifying permeability changes, but both the low frequency and rotating pancake probes clearly identified the changes.

On May 1, 2008, the licensee had the vendor-performed eddy current examinations conducted on the hot leg nozzles. The results definitively showed the existence of Alloy 82/182 nozzle-to-safe-end butt welds. An inspection schedule was established in MRP-139 for these welds to be inspected by December 31, 2009. The corresponding four cold leg nozzle welds are to be completed by December 31, 2010. Since the next scheduled Refueling Outage RFO-25 will be during Fall 2009, the licensee preliminarily indicated that volumetric examinations for all six nozzles would be performed at that time. Plans for subsequent examinations and/or mitigation have not been established.

As noted in Section 1R08.1 above, the inspectors reviewed the complete volumetric examination (ultrasonic testing) records associated with the examinations performed during the Fall 2003, Refueling Outage 21.

.6.01 Licensee's Implementation of the Guidance Document MRP-139 Baseline Inspections

a. The pressurizer and Alloy 82/182 butt welds were replaced with 316 stainless steel during Refueling Outage 23, therefore; MRP-139 baseline inspections were not applicable.

b. At the present time, the licensee is not planning to take any deviations from the baseline inspection requirements of MRP-139, and all other applicable dissimilar metal butt welds are scheduled in accordance with MRP-139 guidelines.

.6.02 Volumetric Examinations

a. No MRP-139 volumetric examinations were performed since Alloy 82/182 dissimilar metal butt welds no longer exist in the pressurizer.

b. No MRP-139 volumetric examinations were performed.

c. No MRP-139 volumetric examinations were performed; thus, there were no personnel qualification records to review.

d. There were no examinations; thus, no deficiencies were identified.

.6.03 Weld Overlays

a. Not applicable
b. Not applicable
c. Not applicable
d. Not applicable

.6.04 Mechanical Stress Improvement

This item is not applicable because the licensee did not employ a mechanical stress improvement process.

.6.05 Inservice inspection program

The licensee's MRP-139 inservice inspection program has not been formally established since the existence of Alloy 82/182 dissimilar butt welds in the reactor vessel nozzle hot and cold legs have recently been confirmed. As mentioned above, preliminary plans include performance of volumetric exams on all six reactor vessel nozzles during the Fall 2009 refueling outage. Additional possibilities include mitigation during the Fall 2009 refueling outage; however, in the absence of mitigation, the licensee would perform a bare metal visual examination of the reactor vessel hot leg nozzles in Spring 2011 and Fall 2012, with volumetric examinations on the two hot leg nozzles and bare metal visual examinations on the four cold leg nozzles in Spring 2014.

Temporary Instruction 2515/172 remains open.

.7 (Closed) NOV 05000285/2007009-01 Failure to Follow Radiation Work Permit Instructions

On May 16, 2007, a notice of violation was issued to the licensee for failure to follow radiation protection procedure and radiation work permit (RWP) instructions. Specifically, one security officer, on at least three occasions, between November 26, 2005, and March 27, 2006, failed to log in on the required RWP and did not activate his electronic alarming dosimeter as required prior to entering the Alpha 1 security post inside a posted radiation controlled area. Corrective actions included: (1) review of radiation survey information for the Alpha 1 post area, (2) removal of the radiation controlled area containing the Alpha 1 post based on review of the radiation data, (3) installation of an area radiation monitor with local and remote readouts and alarm setpoints below the level which would require posting of the area as a radiation area, (4) instructions to the security officers in the event of an alarm on the radiation monitor, and (5) revision of radioactive waste handling procedures when handling highly radioactive materials which could impact the radiation levels at the Alpha 1 security post. On May 1, 2008, corrective actions were reviewed and found to be adequate to prevent recurrence of this issue.

4OA6 Meetings

Exit Meeting Summary

On April 3, 2008, the emergency preparedness inspectors conducted a telephonic exit meeting to present the results of the in-office inspection of the licensee's changes to their emergency plan to Mr. C. Simmons, Supervisor, Emergency Planning, who acknowledged the findings. The inspectors confirmed that proprietary, sensitive, or personal information examined during the inspection had been returned to the identified custodian.

On May 2, 2008, the health physics inspectors presented the occupational radiation safety inspection results to Mr. D. Trausch and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

On May 8, 2008, the reactor inspectors presented the results of this inservice inspection to Mr. D. Bannister, Site Vice President, and other members of licensee management. Licensee management acknowledged the inspection findings.

On June 19, 2008, the inspectors performing the focused baseline inspection of the voiding conditions presented the inspection results to Mr. P. Cronin and other members of the licensee's staff, who acknowledged the findings.

On July 9, 2008, the resident inspectors presented the inspection results to Mr. R. Clemens, Division Manager Nuclear Engineering, and other members of licensee management, who acknowledged the inspection findings. The inspectors confirmed that no proprietary information had been provided.

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, Criterion XVI, states, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment and nonconformance's are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition." Contrary to the above, on February 16, 2007, the licensee inserted auxiliary contacts lubricated with Molykote 55-M grease into Relay 2CR relay for Emergency Diesel Generator 1, a significant condition adverse to quality. Two days earlier, on February 14, 2007, these same contacts had failed due to the application of Molkote 55-M rendering the Diesel Generator 1 inoperable and resulting in NRC Violation 05000285/2007011-03. The reintroduction of the unapproved lubricant was discovered by the licensee through a review of completed work orders. This finding only had very low safety significance because the diesel generator was subsequently verified to be operable. This finding was identified in the licensee's corrective action program as CR 2008-0071.

- Technical Specification 5.8.1.a states, in part, "Written procedures shall be established, implemented, and maintained covering the following activities. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978." Regulatory Guide 1.33, Revision 2, Appendix A, 1978, Section 3.f(1), requires procedures for maintaining containment integrity. Contrary to the above, on April 24, 2008, the licensee violated the containment integrity requirements of Procedure OI-CO-4, "Containment Closure," Revision 45. Specifically, the reactor coolant system level was lowered to reduced inventory (a condition which required containment integrity) while two containment isolation Valves HCV-401C and HCV-403C were removed from the system. This condition was identified by the licensee during a routine walkdown of the spaces. This finding was considered to have very low safety significance because of the small size of the opening. This finding was identified in the licensee's corrective action program as CR 2008-2706.

- Technical Specification 2.6 states, in part, "Containment integrity shall not be violated unless the reactor is in a cold or refueling shutdown condition. Without containment integrity, restore containment integrity within one hour or be in at least hot shutdown within the next 6 hours." Contrary to the above, from March 15, 2008, at 08:33 a.m., until March 16, 2008, at 2:01 p.m., containment integrity did not exist due to a leaking relief valve. This condition initially went unrecognized by operators, but was subsequently identified by the licensee. This finding was considered to have very low safety significance because of the small size of the opening (2-inch relief valve) which was connected to a closed system. This finding was identified in the licensee's corrective action program as CR 2008-1622 and was reported as LER 05000285/2008002-00.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Anderson, Supervisor, Radwaste
M. Anielak, Manager of Shift Operations
D. Bannister, Vice President
S. Baughn, Supervisor, Reactor Performance Analysis
P. Christensen, Sr. Technician, Radiation Protection
A. Clark, Manger, Security
R. Clemens, Division Manager, Nuclear Engineering
M. Cooper, Compliance Engineer, Certrec
P. Cronin, Manager, Operations
H. Faulhaber, Division Manager, Nuclear Asset Management
M. Frans, Manager System Engineering
J. Gasper, Acting Manager, Design Engineering
T. Giebelhausen, Supervisor, Nuclear Training/Nuclear Support
W. Goodell, Division Manger, Quality and Performance Improvement
D. Guinn, Licensing Engineering
P. Hamer, Inservice Inspection Program Engineer
R. Haug, Manager, Radiation Protection
J. Herman, Manager, Engineering Program
T. Hutchinson, Steam Generator Program Engineer
R. Johansen, Manager, Maintenance
P. Kellog, Sr. Technician, Radiation Protection
D. Little, Specialist, Radiation Health
C. Longua, Control Room Supervisor
T. Maine, Supervisor, Radiation Protection ALARA
O. Manager, Nuclear Procurement Services
T. Matthews, Manager, Nuclear Licensing
E. Matzke, Licensing
J. McManis, Manager, Licensing
T. Mitchell, Component Engineering
T. Nellenbach, Division Manager, Nuclear Operations/Plant Manager
T. Pilmaier, Manager, Performance
B. Ricks, Reactor Operator
S. Shea, Shift Technical Advisor
J. Shuck, Training
C. Simmons, Supervisor, Emergency Planning
D. Spires, Manager, Integrated Work Management
T. Steckelberg, Sr. Technician, Radiation Protection
T. Stella, Shift Manager
D. Sweeney, Senior Reactor Operator
M. Tesar, Division Manager, Nuclear Support
D. Trausch, Assistant Plant Manager
T. Uehling, Manager, Chemistry
R. Westcott, Manager, Quality

NRC Personnel

J. Hanna, Senior Resident Inspectors, RIV
J. Kirkland, Resident Inspectors
M. Runyan, Senior Reactor Analyst, RIV

Other Personnel

K. Saltzman, Authorized Nuclear Inservice Inspectors, Hartford Steam Boiler and Insurance Company

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000285/2008003-01	NCV	Inadequate Boric Acid Corrosion Control Procedure (Section 1RO8.3)
05000285/2008003-02	NCV	Failure to Conspicuously Post a Radiation Area (Section 2OS1)
05000285/2008003-03	NCV	Failure to Follow Procedures (Section 2OS2)
05000285/2008003-04	NCV	Failure to Write an Adequate Shutdown Procedure (Section 4OA3)
05000285/2008003-05	NCV	Inadequate Corrective Actions for a Containment Spray Design Deficiency (Section 4OA5.3)

Closed

05000285/2007009-01	NOV	Failure to Follow Radiation Work Permit Instructions (Section 4OA5.7)
05000285/2007011-03	NOV	Failure to Provide Procedure for Safety Related Maintenance Activities (Section 4OA2.1)
05000285/2008001-00	LER	Reactor Trip Due to Turbine Control System Failure (Section 4OA3.2)
05000285/2008002-00	LER	Loss of Containment Integrity Due to a Leaking Isolation Valve (Section 4OA3.3)

05000285/2008006-01	URI	High Contact Resistance on Main Steam Bypass Valve Relay Contacts (Section 4OA5.2)
05000285/2008006-02	URI	Containment Cooling Design Requirements and Licensing Review (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedure FCSG-15-24, “Housekeeping,” Revision 5

Procedure FCSG-1, “Duty Assignments,” Revision 7

Procedure AOP-1, “Acts of Nature,” Revision 23

Procedure AOP-31, “161 KV Grid Malfunctions,” Revision 7

Procedure NOD-QP-36, “Grid Operations and Control of Switchyard at FCS,” Revision 15

Procedure OI-EG-3, “EMS Post-FCS-Trip 161 KV Voltage Prediction and Switchyard Status,” Revision 5

Control Room Operator logs dated May 15, 2008

Condition Report 200602454

Condition Report 200603650

Condition Report 2007-3760

Section 1RO5: Fire Protection

Standing Order SO-G-28, “Station Fire Plan,” Revision 71

Standing Order SO-G-91, “Control and Transportation of Combustible Materials,” Revision 24

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

Abnormal Operating Procedure AOP-6, “Fire Emergency,” Revisions 20 and 21

EA-FC-97-001, “Fire Hazards Analysis (FHA) Manual,” Revision 14

USAR, Section 9.11, “Fire Protection Systems,” Revision 17

<u>Section 1R08: Inservice Inspection Activities</u>

<u>Condition Reports</u>

2008-0990	200600508	2008-2495	200605173
200700339	2008-1782	200700553	2008-2235
2008-1799	2008-0911	2008-3060	200603555
200501092	200501093	200501094	2008-3099
200603617	2007-4813	2008-2567	2008-1891

<u>Procedures</u>

QCP-400, Visual Inspection, Revision 11

QCP-200, Certification Requirements for Quality Control Inspectors, Revision 32

SE-EQT-MX-0002, Carbon Steel and Alloy Steel Fasteners Inservice Testing Inspections, Revision 9

PBD-10, Boric Acid Corrosion Prevention, Revision 11

OPPD-UT-CP-2, Procedure for Inspection System Performance Checks, Revision 1

OPPD-VT-98-1, Visual Examination: VT-1, Revision 1

OPPD-VT-98-3, Visual Examination for Mechanical and Structural Condition of Components and Their Supports, Revision 1

OPPD-PT-98-1, Liquid Penetrant Examination – Solvent Removable, Visible Dye Technique, Revision 3

OPPD-UT-98-5, Ultrasonic Examination of Studs/Bolts Greater than Two Inches in Diameter, Revision 2

OPPD-UT-98-1, Manual Ultrasonic Examination of Ferritic Piping Welds, Revision 2

PDI-UT-1, Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds, Revision C

OPPD-UT-98-2, Manual Ultrasonic Examination of Austenitic Piping Welds, Revision 2

PDI-UT-2, Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds, Revision C

SO-R-2, Condition Reporting and Corrective Action, Revision 39

SO-M-101, Maintenance Work Control, Revision 75

SE-ST-SDC-3003, Shutdown Cooling Suction Header Refueling Leakage Test, Revision 20

QC-ST-MX-3001, VT-2 Examination of Normally Insulated Class 1 Pressure Retaining Bolted Connections in Systems Borated for Reactivity Control, Revision 3

QC-ST-MX-3002, VT-2 Examination of Normally Insulated Class 2 and 3 Pressure Retaining Bolted Connections in Systems Borated for Reactivity Control, Revision 4

WDI-Q&FT – 1024, Material Characterization along DMW on RPV Nozzle Safe-End Qualification Test Report, Revision 0

<u>FORT CALHOUN STATION Steam Generator Tube Examination Technique Specification Sheets and Qualifying EPRI ETSSs</u>

FTC 1-08-BIB-A-D and FTC 1-BOB: 96004.2 R11, 27091.2 R0, 96004.1 R11, 24013.1 R2, 96010.1 R7

FTC 1-08-R1-A-C, R2-A-B, R3-A-C: 27901.1 R0, 27902.1 R0, 27903.1 R0, 27904.1 R0, 27905.1 R0, 27906.1 R0, 27907.1 R0, and 96910.1, R10

<u>Work Orders</u>

WO 248791-01 WO 266747-01 WO 246229 WO 273312

<u>Miscellaneous</u>

Wesdyne International, Material Characterization on Outlet Nozzles DMW at Fort Calhoun Station, 2008 Outage

SG-CDME-08-4, Fort Calhoun Station Steam Generator 08RFO Degradation Assessment Report, Revision 1

ISI Program Plan, Omaha Public Power District, Fort Calhoun Station Fourth Interval Inservice Inspection Ten Year Program Plan (2003-2013), Revision 2

MRS-TRC-1881, Use of Appendix H Qualified Techniques at Fort Calhoun Station for the Spring 2008 S/G Inspection, dated 3/25/08

<u>Calculations</u>

FC 07361, Calculation of Effective Degradation Years (EDY) for the FORT CALHOUN STATION Reactor Pressure Vessel Head (RPVH) During Cycle 24, dated 4/25/08

<u>Certified Material Test Reports</u>

Magnaflux Spotcheck Cleaner, Batch 03F02K
Magnaflux Spotcheck Penetrant, Batch 06H10K
Magnaflux Spotcheck Developer, Batch 03J03K
Welding Material ER 80S-B2, SFA 5.28, HT XA 8361, dated 6/28/05

Drawings

SQUID in Nozzle – SUPREEM™ eQuivalent Safe-end Ultrasonic Inspection Device

Quality Control Inspection Reports

20080130	20080150	20080116
20080110	20060270	20060601

Presentation

PA-MSC-0298, RV Primary Nozzle Weld Inlay PDI Equivalency Testing and Process Qualification, Revision 0

Boric Acid Evaluations

CR 200700525	CR 2007-3802	CR 2007-3801	CR 2007-3795
CR 200602767	CR 200605471	CR 200600508	CR 2008-0990
CR 2008-1605	CR 2008-1602	CR 2008-1603	CR 2008-1606
CR 2008-1604			

Requests for Relief

RR-8, Use of Alternative to Appendix VIII, Supplement 10
RR-9, Use of Alternative to Appendix VIII, Supplement 14

Inservice Inspection Code Cases

N-460	N-461-1	N-533-1	N-566-1
N-624	N-648-1	N-663	N-623

Section 1R11: Licensed Operator Requalification Program

Simulator package for June 18, 2008
Open Simulator Discrepancy Reports (All)
Current Simulator Differences List
Simulator Modification Procedures
Verification and Validation Procedures
Current operator license list from Fort Calhoun Station
SO-O-21, "Conduct of Operations," Revision 76
Condition Report 2008-4490

Section 1R12: Maintenance Effectiveness

PED-SEI-34, "Maintenance Rule Program," Revision 6
MRII-0, "General Instructions," Revision 6
MRII-1.1, "Scoping," Revision 2
MRII-1.2, "Risk Significance Determination," Revision 5
MRII-2, "Setting Performance Criteria," Revision 4

MRII-2.1, “Monitoring and Reporting of SSC Availability,” Revision 4
MRII-6, “Placement of SSC’s into Category (a) (1) or (a) (2),” Revision 8
MRII-7.1, “Periodic Assessment,” Revision 3

Condition Reports

2008-1398	200400880	2007-4321

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

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

Risk evaluation and risk management actions per e-mail from Alan Hackerott, OPPD dated May 12, 2008

FCSG-19, Attachment 2, “EOOS Operating Instructions,” Revision 8

NOD-QP-36, “Grid Operations and Control of Switchyard at FCS,” Revision 15

Standing Order SO-O-21, “Shutdown Operations Protection Plan,” Revision 30

Standing Order SO-M-100, “Conduct of Maintenance,” Revision 48.

Section 1R15: Operability Evaluations

Condition Reports

1996-0062	1996-0544	1996-1446	1996-1519	2003-5295
2008-1089	2008-1666	2008-1683	2008-2304	2008-2586
2001-1437	2008-0080	2008-0148	2008-0326	2008-0336
2008-0958	2008-3316			

Miscellaneous

Work Order 00216102-01

EC 42990

Procedure SE-PM-CH-0202, “Chemical and Volume Control System CH-202 Performance Test,” Revisions 1 and 2

EOP/AOP Attachment 9, “Simultaneous Hot and Cold Leg Injection,” Revisions 24 and 25

EOP/AOP Attachment 10, “Simultaneous Hot and Cold Leg Injection Without Instrument Air,” Revision 25

EOP/AOP Attachment 11, “Alternate Hot Leg Injection,” Revisions 24 and 25

EOP/AOP Attachment 26, “Total SI Pump Flow to Match Decay Heat vs. Time After Trip,” Revision 25

NCR Information Notice 93-66, “Switchover to Hot-Leg Injection Following a Loss-of-Coolant Accident in Pressurized Water Reactors,” August 16, 1993

M. Gargallo et el., 2004, “Counter-current flow limitations during hot leg injection in pressurized water reactors,” Science Direct (2004)

NRC Memorandum from Nakoski to Terao, “Large Break LOCA Safety Evaluation Fort Calhoun License Amendment Request,” July 26, 2006

Westinghouse Letter LTR-LIS-06-486 [C. H. Boyd] to OPPD, “Fort Calhoun Unit 1 Evaluation of RCS Change Impact on Post-LOCA,” August 29, 2006

Safety Analysis for Operability 2008-0080, “Compensatory Measures for Simultaneous Hot and Cold Leg Injection to Address CH-202 Bypass Flow,” January 10, 2008

Root Cause Analysis Report 2008-0080, “Reliance on CH-202 as a Boundary for Simultaneous
Hot/Cold Leg Injection,” February 12, 2008

Westinghouse Letter LTR-LIS-08-163 [Gates] to OPPD, “Fort Calhoun Minimum Required Flow Times of 5.5, 12, 18, and 24 Hours,” February 20, 2008

Proto-Power Corporation Letter 018FCH/051508/L08001 [D’Angelo] to OPPD [Friedman], “PROTO-FLO Output Reports Pertaining to Flow through CH-202 Should HCV-247 Fail Open During Hot Leg Injection with Degraded Pumps to Support Past Operability Analysis,” February 26, 2008

Westinghouse Letter CFTC-08-12 [Rajan] to OPPD [Swearingn], “Consultation Support Related to SIS Hot Leg Injection System Alignment,” March 7, 2008

Design Basis Document SDBD-CH-108, “Chemical and Volume Control Systems,” Revision 20

Design Basis Document SDBD-HP-132, “High Pressure Safety Injection,” Revision 20

Updated Safety Analysis Report Section 14.15, “Loss-Of-Coolant Accident,” Revision 26

Calculation FC05584, “Effect on CH-202 of HCV-247 Failing Open During Hot Leg Injection,” Revision 0

Calculation FC07078, “Recirculation Phase System Performance for Safety Injection and Containment Spray Systems,” Revision 0

Calculation EA-FC-90-109, “Function of CH-202/CH-345,” Revision 0

Westinghouse Calculation CN-LIS-08-33, “Fort Calhoun SIS Injection Alignment Required Flows,” Revision 0

Proto-Power Corporation Calculation 08-025, “Flow through CH-202 Should HCV-247 Fail Open During Hot Leg Injection,” Revision A

Drawing E-23866-210-110 Sheet 1, “Reactor Coolant System Flow Diagram P&ID, Revision 80

Drawing E-23866-210-120 Sheet 1A, “Chemical and Volume Control System P&ID,” Revision 19

Drawing E-2866-210-130 Sheet COV, “Composite Flow Diagram Safety Inection and Containment Spray System P&ID,” Revision 39

Regulatory Guide 1.29, “Seismic Design Classification,” Revision 4

Information Notice 2000-20, “Potential Loss of Redundant Safety-Related Equipment Because of the Lack of High-Energy Line Break Barriers,” December 11, 2000

NUREG 0800 Section 3.6.1, “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” Revision 3

NUREG 0800 Branch Technical Position 3-3, “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” Revision 3

Operability Evaluation 2008-2304, April 16, 2008

Operability Evaluation 2008-2304, April 18, 2008

Operability Evaluation 2008-2586, April 26, 2008

EA-FC-91-031, “Potential Failures of Auxiliary Steam Piping and the Possible Effects on the Operability of Vital Equipment,” Revision 1

EA-FC-92-027, “Component Cooling Water and Raw Water Post-Accident Single Failure Evaluation,” Revision 3

EA-FC-93-085, Preferred Safe Shutdown Path for Fort Calhoun Station, Revision 5

NRC Letter 72-007 from AEC [Giambusso] to OPPD [Wilkins], “General Information Required for Consideration of the Effects of a Piping System Break Outside Containment,” December 14, 1972

OPPD Letter LIC 73-007 from OPPD [Wilkins] to AEC [Giambusso], “Initial Phase Postulated High Energy Line Rupture Outside the Containment,” March 14, 1973

OPPD Letter LIC 73-012 from OPPD [Wilkins] to AEC [Giambusso], “Postulated High Energy Line Rupture Outside the Containment,” May 15, 1973

NRC Letter 73-0029 from AEC [O’Leary] to OPPD [Wilkins], “Criteria for Determination of Postulated Break and Leakage Locations in High and Moderate Energy Fluid Piping Systems Outside of Containment Structures,” July 12, 1973

Section 1R18: Plant Modifications

Temporary Modification Number EC43192
Condition Report 2008-2937
Work Order 277777-22
LIC-07-0054, Uprate of Shutdown Cooling System Entry Conditions LAR, 10/12/07
License Amendment No. 256, issued 5/29/08
EC 35639, Shutdown Cooling Entry Conditions Uprate, Revision 0
EC 32388, Feedwater Digital Control System Modification

Section 1R19: Postmaintenance Testing

Work Orders

00285068-01	00298858-01
00300156-01	00306323-06
00266243-01	

Procedures

EM-ST-CEA-0001, Refueling CEA Position Indication Check, Revision 3

OP-ST-RW-3031, "AC-10D Raw Water Pump Quarterly Inservice Test," Revision 31

IC-ST-ESF-0001, "Functional Test of Pressurizer Pressure Low Signal (PPLS) Actuation and Blocking Logic," Revision 10

Section 1R20: Refueling and Other Outage Activities

Shutdown Safety Advisor's Log dated May 20, 2008
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 14
Licensee Response Letter to NRC Generic Letter 88-17

Section 1R22: Surveillance Testing

Surveillance Test SE-ST-MS-3005, "Main Steam Safety Valves Set Pressure Testing Using Furmanite's Trevitest Equipment," Revision 5

Functional Test EM-FT-EX-0200, "Functional Test of Auto Start Prohibit and Undervoltage Trip

Prohibit of 480 and 4160 Volt Breakers," Revision 6

Work Order 00261465-01

OP-ST-SI-3003, "Low Pressure Safety Injection and Containment Spray System Pump and Check Valve Test," Revision 19

Root Cause Analysis “Valves LCV-383-1 and 2 Exceeded Surveillance Test Leakage Criteria of Technical Specification 3.16(2)a”

Fisher Controls Drawing 12B7109, “Valve Body, ANSI Class 150, Wafer Style,” Revision E

SE-ST-SI-3027, “RHR Headers ‘A’ and ‘B’ Refueling Hydrostatic and Leakage Test,” Revision 15

SE-ST-SDC-3002, “Shutdown Cooling Pump Refueling Leakage Test,” Revision 6

SE-ST-SI-3005, “Measurement of Post RAS Leakage Test to the Safety Injection Refueling Water Tank (SIRWT),” Revision 20

Data sheets showing historical leakage from Emergency Core Cooling Systems, dated April 20, 2005, October 9, 2003, May 27, 2002, April 23, 2001

SE-ST-ILRT-0001, "Containment Integrated Leakage Rate Test (CILRT)," Revision 5

Condition Report 2008-2919

<u>Sections 2OS1: Access Controls to Radiologically Significant Areas and 2OS2: ALARA Planning and Controls</u>

<u>Corrective Action Documents</u>

2008-0021	2008-0120	2008-0133	2008-0186	2008-0523
2008-0526	2008-0856	2008-0860	2008-1168	2008-1681
2008-1851	2008-1886	2008-2174	2008-2193	2008-2402
2008-2458	2008-2507	2008-2621	2008-2680	2008-2770
2008-2790	2008-2879	2008-2904		

<u>Audits and Self-Assessments</u>

Self-Assessment 07-20
Quality Surveillance Observation 1, dated 1/07/08
Quality Surveillance Observation 3, dated 1/04/08
Quality Surveillance Observation 71, dated 3/17/08
Quality Surveillance Observation 75, dated 3/17/08
Quality Surveillance Observation 76, dated 3/18/08
Quality Surveillance Observation 77, dated 3/18/08

<u>Radiation Work Permits</u>

08-3302, Regulatory Tours
08-0200, Routine Decontamination Duties
08-2515, Ventilation Maintenance and Inspections
08-3512, Reactor Head Disassembly and Transport

Procedures

RPI-1, Personnel Monitoring and Decontamination, Revision 13
RP-202, Radiological Surveys, Revision 31
RP-203, Air Sample Collection and Analysis, Revision 17
RP-204, Radiological Area Controls, Revision 49
RP-212, Diving Operations Within Radiologically Controlled Areas, Revision 8
RP-608, Dose Calculations from Contamination, Revision 13
SO-O-47, Spent Fuel Pool Inventory Control, Revision 7
SO-R-2, Condition Reporting and Corrective Action, Revision 39

Miscellaneous

2007 Personnel Contamination Log
2008 Personnel Contamination Log
Exposure Evaluation Reports; 07-014, 08-008, 08-011

Section 4OA1: Performance Indicator Verification

Procedures

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

Section 4OA2: Identification and Resolution of Problems

Fort Calhoun Record of Telephone Communication RTC-08-015

EM-PM-EX-1100, "480 Volt Motor Control Center Maintenance," Revision 20

MM-ST-DG-0001, "Diesel Generator DG-1 Inspection," Revision 65

Fort Calhoun White Paper, "Basis for Eliminating Binding of DG-1 2CR Relay Auxiliary Contact Assembly Actuator as a Causal Factor for High Contact Resistance"

Fort Calhoun Equipment Reliability Optimization Project Status Report, dated May 30, 2008

Section 4OA3: Event Follow-Up

Root Cause Analysis, "Missed Technical Specification Entry for Loss of Containment Integrity," Revision 1

Condition Reports:

2008-1622 2008-2630 2008-2615 2008-3506 2008-3502

Control Room Operating Logs dated May 20, 2008

Event Notification #44228 dated May 21, 2008

Figure 8.1-1, "Simplified One Line Diagram Plant Electrical System," Revision 128

Design Basis Document SDBD-SI-130, “Shutdown Cooling,” Revision 19

Design Basis Document SDBD-SI-133, “Low Pressure Safety Injection System,” Revision 26

USAR Section 9.3, “Shutdown Cooling System,” Revision 11

OP-3A, “Plant Shutdown,” Revision 73

OI-RC-4A, “Pressurizer Cooldown and Venting,” Revision 24
TDB-III.28, “Reactor Vessel Level Monitoring System,” Revision 8

Plots of RCS Parameters (pressures, temperatures, levels, flows) dated 6/10/08 – 6/11/08

Crew statements from the Shift Manager, Control Room Supervisor, Reactor Operator, and the second Senior Reactor Operator in the Control Room

Condition Report 20084131, dated 6/12/08

Section 4OA5: Other Activities

Condition Reports

2003-4445	2003-4581	2004-1834	2005-5841	2006-4062
2006-4333	2006-5289	2006-5990	2008-1586	2008-2791
2008-3000	2008-3235			

Section 4OA6: TI-166 Documents Reviewed

SO-M-101/EC30663, Standing Order Maintenance Work Control/GSI-191 Implementation, Revision 0

SO-O-25/EC43026, Standing Order/Temporary Modification Control, Revision 69

EC 38570, Removal and Replacement of Trisodium Phosphate, Revision 14

EC 40070, Revision 0

EC 38581, 2006 RFO Partial Containment Strainer Modules, September 6, 2006

LIC-08-0021, Supplemental Response to Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors”, February 29, 2008

LIC-06-0067, “Revised Request for an Extension to the Completion Date for Corrective Actions Taken in Response to Generic Letter 2004-02, June 9, 2005

LIC-05-0017, 90 Day Response to Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors”, March 4, 2005

LIC-05-0101, Follow-up Response to Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors”, August 31, 2005

LIC-05-0131, Request for an Extension to the Completion Date for Corrective Actions Taken in Response to Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors”, November 18, 2005

LIC-07-0052, Ft. Calhoun Station Unit No. 1 License Amendment Request (LAR), “Modification of the Containment Spray System Actuation Logic”

Section 4OA7: Licensee-Identified Violations

Condition Reports:

2008-0071 2008-2828 2008-1622 2008-2615

List of Criticality Level N-1 Equipment Using CR-105X Contacts

Operability Determination for Condition Report 2008-0071

Root Cause Assessment “Loss of Containment Closure during Reduced Inventory,” dated June 12, 2008

LIST OF ACRONYMS

CAP	corrective action program
CFR	Code of Federal Regulations
CR	Condition Report
EPRI	Electric Power Research Institute
NCPM	net counts per minute
NCV	noncited violation
NDE	nondestructive examination
NRC	Nuclear Regulatory Commission
OP	Operating Procedure
PT	penetrant test
RCP	reactor coolant pumps
RCS	reactor coolant system
RVLMS	reactor vessel level management system
RWP	radiation work permit
SDC	shutdown cooling
SGs	steam generators
SIRWT	safety injection and refueling water storage tank
SSC	structure systems and components
TS	technical specification
USAR	Updated Safety Analysis Report
UT	ultrasonic testing
VT-2	visual test