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

August 5, 2005

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

SUBJECT: FORT CALHOUN STATION - NRC INTEGRATED INSPECTION REPORT 05000285/2005003

Dear Mr. Ridenoure:

On June 30, 2005, 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 6, 2005, with Mr. David Bannister, Plant Manager 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 two NRC-identified and four self-revealing findings that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these issues. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in Section 4OA7 of this report. The NCVs are described in the subject inspection report. If you contest the violations or significance of the NCV's, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Fort Calhoun Station facility.

In accordance with 10 CFR 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).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

David N. Graves, Chief
Project Branch E
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2005003
w/attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285
License:	DPR-40
Report:	05000285/2005003
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, 2005
Inspectors:	J. Hanna, Senior Resident Inspector L. Willoughby, Resident Inspector B. Baca, Health Physics Inspector J. Tapia, P.E., Senior Reactor Inspector, Engineering Branch J. C. Kirkland, Project Engineer, Project Branch E R. Lantz, Senior Emergency Preparedness Inspector D. Carter, Health Physics Inspector G. Pick, Senior Reactor Inspector, Engineering Branch 2
Approved By:	David N. Graves, Chief, Project Branch E Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000285/2005003; 04/01/2005 - 06/30/2005; Fort Calhoun Station, Integrated Resident and Regional Report and Occupational Radiation Safety.

The report covered a 3-month period of inspection by resident inspectors and announced inspections by a regional health physicist inspector, reactor inspector, project engineer, and emergency preparedness inspector. Six Green noncited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. A self-revealing noncited violation was identified as a result of the failure of the refueling machine operators to follow the procedure for transferring fuel in the reactor vessel as required by Technical Specification 5.8.1.a. This failure resulted in not identifying that fuel assembly Y019 was improperly seated into core location H17 until the adjacent fuel assembly was loaded and properly seated. This finding had crosscutting aspects associated with human performance in that the operators failed to follow procedures as required. This violation was entered into the licensee's corrective action program as CR 200502434.

  This finding was more than minor since it is associated with the fuel cladding human performance attribute of the cornerstone. The finding was characterized by regional management as having very low safety significance because there was no damage to fuel pins or the fuel assembly (Section 1R20).

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," was identified because the licensee failed to follow the procedure for ensuring that an outside contractor was properly qualified to perform safety significant activities under Omaha Public Power District's established quality assurance plan. Specifically, the licensee failed to review and approve the R. Brooks and Associates, Inc., eddy-current testing personnel certifications, equipment calibrations and procedures prior to performing work. This finding had human performance crosscutting aspects regarding failure to follow procedures.

  The finding was greater than minor because it was associated with the performance attribute of the barrier integrity cornerstone and impacted the cornerstone objective of providing reasonable assurance that physical design barriers, in this case the reactor vessel, protect the public from radionuclide releases caused by accidents or events (Section 4OA5).

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion IX, “Control of Special Processes,” was identified because the licensee failed to adequately certify their nondestructive testing personnel in accordance with the American Society of Nondestructive Testing, “Standard for Qualification and Certification of Nondestructive Testing Personnel,” CP-189-1991. This finding had human performance crosscutting aspects regarding failure to follow nondestructive testing personnel certification procedures.

  The finding was greater than minor because it was associated with the performance attribute of the barrier integrity cornerstone and impacted the cornerstone objective of providing reasonable assurance that physical design barriers, in this case the reactor vessel, protect the public from radionuclide releases caused by accidents or events (Section 4OA5).

Cornerstone: Occupational Radiation Safety

- Green. A self-revealing noncited violation of 10 CFR Part 50 Appendix B, Criterion V, was identified based on the licensee’s operational procedure for containment building ventilation being inadequate. Specifically, the procedure that controlled the containment ventilation fans did not state the order to start the supply and exhaust fans, resulting in contamination of the auxiliary building. This finding had human performance crosscutting aspects in that the subject procedure was inadequate. This finding was also entered into the licensee’s corrective action program as CR 200501394.

  The performance deficiency was an inadequate containment building ventilation system operational procedure. This finding was more than minor because it affected the Occupational Radiation Safety cornerstone objective to protect worker health and safety from radiation and radioactive materials. Specifically, the Plant Facilities/Equipment Attribute of the cornerstone was affected and involved unplanned and unintended dose to workers. The issue screened out as Green 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 condition has been entered into the licensee’s corrective action program (Section 1R14).

- Green. A self-revealing noncited violation was reviewed for the failure to comply with a radiation work permit requirement. Specifically on March 3, 2005, a job supervisor and a worker did not notify radiation protection of a tool change, from a band saw to a grinder, as required by the radiation work permit. Contamination levels were as high as 500 milirad per hour per 100 centimeters square. As a result, several individuals participating in the work activity became contaminated and alarmed the personnel contamination monitors upon exiting the radiologically controlled area. Four individuals had low levels of internal contamination. The maximum dose assigned was 37 millirem. This finding had a crosscutting aspect with respect to human performance because the job supervisor or worker did not inform radiation protection before making a change in approved cutting instruments which directly contributed to the finding.

The finding was greater than minor because it was associated with the Occupational Radiation Safety attribute of Program and Process and affected the cornerstone objective. The failure to comply with a radiation work permit requirement resulted in the low-level internal contamination of four workers. The finding was determined to be 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 was placed into the licensee's corrective action program as Condition Report 2005-0943 (Section 2OS2).

- Green. A self-revealing noncited violation was reviewed for the failure to comply with a Technical Specification required radiation work procedure. Specifically on April 7, 2005, an individual performing work in a high radiation area received a dose rate alarm and did not notify radiation protection personnel. This finding had a crosscutting aspect with respect to human performance because the worker did not notify radiation protection personnel of a dose rate alarm in a high radiation area which directly contributed to the finding.

  The finding was greater than minor because it was associated with the Occupational Radiation Safety attribute of Program and Process and affects the cornerstone objective. The failure to comply with a radiation work procedure could result in an increase in a personnel dose. The finding was determined to be 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 was placed into the licensee's corrective action program as Condition Report 2005-1912 (Section 2OS2).

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and the associated corrective actions are listed in (Section 4OA7).

REPORT DETAILS

Summary of Plant Status

At the start of the inspection period, the plant was in Mode 5 with all fuel off-loaded to the spent fuel pool for the spring 2005 refueling outage. The reactor was restarted on May 30, 2005, and was synchronized to the electrical grid on June 1. On June 4 the reactor was manually shut down to repair degraded seals on Reactor Coolant Pumps A and B. On June 13 following repairs to the Reactor Coolant Pumps, the reactor was restarted and synchronized to the electrical grid. On June 17 reactor power was reduced from 98 percent to 72 percent to perform repairs on a leaking high pressure feedwater heater. Power was increased to 98 percent following the feedwater heater repairs. On June 21 and 23 power was reduced to 52 percent and 29 percent, respectively, due to ongoing condenser tube leakage. On June 24 reactor power was increased to 86 percent once secondary chemistry parameters were again within allowed limits. At the end of the inspection period the plant was at 96 percent reactor power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspector reviewed the licensee's high wind weather protection requirements (one inspection sample) utilized during tornado season. The inspector reviewed Procedure FCSG-1, "Duty Assignments," Revision 3, and performed a walkdown of accessible outside areas to identify potential missile hazards. The inspector verified that tours were conducted in accordance with the procedure.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

Partial Equipment Walkdowns. The inspectors performed partial walkdowns of the following trains of equipment during outages, operation, or testing of redundant trains (three inspection samples). The inspectors verified that the following systems were properly aligned in accordance with system piping and instrumentation drawings and plant procedures:

- Emergency Diesel Generator 1 while Emergency Diesel Generator 2 was out of service for surveillance testing

- Motor-driven auxiliary feedwater pump and turbine-driven auxiliary feedwater pump while the diesel-driven auxiliary feedwater pump was out of service for maintenance

- Diesel Generator 1 jacket water system while Diesel Generator 2 was inoperable for surveillance testing

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors performed routine fire inspection tours (six inspection samples) and reviewed relevant records for plant areas important to reactor safety. The inspectors observed the material condition of plant fire protection equipment, the control of transient combustibles, and the operational status of barriers. The inspectors compared in-plant observations with commitments in the licensee's Updated Fire Hazards Analysis Report. The following fire areas were inspected:

- Fire Area 32 - Air Compressor Area (Room 19)
- Fire Area 20.1 - East Personnel Corridor 26 and PAL Area (Corridor 26)
- Fire Area 1 - Safety Injection and Containment Spray Pump Area 1 (Room 21)
- Fire Area 41 - Cable Spreading Room (Room 70)
- Fire Area 20.7 - New Fuel Storage and Uncrating Room (Room 25)
- Fire Area 20.6 - Drumming Room (Room 27)

b. Findings

No findings of significance were identified

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the design basis flood heights, as specified in the Updated Safety Analysis Report, and the equipment found outside of the auxiliary and turbine building structures for susceptibility to external flooding events (one inspection sample). Specifically, the inspectors performed walkdowns of FO-1 and FO-10 Diesel Fuel Oil Storage Tanks to verify that the equipment was not subject to damage or water infiltration as a result of external flooding. The inspectors also reviewed the Probabilistic

Risk Assessment Summary Notebook; Procedure AOP-01, "Acts of Nature," Revision 16; and Procedure PE-RR-AE-1001, "Floodgate Installation and Removal," Revision 1. The inspectors verified that the licensee's flood mitigation plans and equipment were consistent with design basis requirements.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors observed the state of cleanliness of Component Cooling Heat Exchanger AC-1A when it was disassembled to remove a tube for microbiological assessment (one inspection sample.) The inspectors reviewed the latest test acceptance criteria and results to ensure differences between testing conditions and design conditions were considered. In addition, the inspectors reviewed the surveillance test against industry recommendations to ensure the surveillance test was adequate for identifying negative performance trends. The inspectors also reviewed the pre-established engineered acceptance criteria to verify heat exchanger and system operability.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections. The inspection procedure requires review of two or three types of nondestructive examination activities (volumetric, surface, and visual). The inspector reviewed multiple examples of all three types.

The procedure requires review of one or two examinations from the previous outage with recordable indications that were accepted for continued service. There were no recordable indications accepted for continued service identified during the previous outage.

If the licensee completed welding on the pressure boundary for Class 1 or 2 systems since the beginning of the previous outage, then the procedure requires verification for one-to-three welds that acceptance and preservice examinations were done in accordance with American Society of Mechanical Engineers (ASME) Code. There was no welding on Class 1 or 2 systems performed since the beginning of the previous outage.

The procedure requires verification that one or two ASME Section XI Code repairs or replacements meet Code requirements. The inspectors verified four Section XI replacements (Safety Injection Valves 123, 185, 188, 220, and Safety Injection Piping 2501R).

The inspector verified, through direct observation or record review, that ultrasonic, eddy-current, liquid penetrant, magnetic particle or visual examinations of the systems/components below were performed in accordance with ASME Code requirements.

System	Component/Weld Identification	Examination Method
Reactor Coolant	Pressurizer Lower Girth (shell to head) Weld PRZ-SC-3-403	Ultrasonic
Feedwater	Steam Generator A Feedwater Nozzle Weld 16-FW-2001/12	Magnetic Particle
Reactor Coolant	Reactor Pressure Vessel Closure Head Welds RPVCH-CRD-BO-41 & -41-2.	Liquid Penetrant
Auxiliary Cooling Water	Trapeze Strut 8-AC-2003/01-PR	Visual
Feedwater	Steam Generator A Extension Ring to Shell Weld SG-1-4B	Ultrasonic
Component Cooling Water	Shutdown Cooling Heat Exchanger AC-4B	Eddy Current
Feedwater	Steam Generator A Lower Head to Extension Ring Weld SG-1-C-2	Ultrasonic

During the review of each examination, the inspector verified that the correct nondestructive examination procedures were used, that examinations and conditions were as specified in the procedure, and that test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspector reviewed documentation to verify that indications revealed by the examinations were dispositioned in accordance with the ASME Code specified acceptance standards. The nondestructive examination certifications of the personnel observed performing examinations or identified during review of completed examination packages were also verified.

Steam Generator Tube Inspection Activities. The inspector verified that the steam generator tube eddy-current examination scope and expansion criteria met the Technical Specification requirements, industry guidelines, and commitments made to the NRC and confirmed that known areas of potential degradation based on site-specific and industry experience were included in the scope of the inspection. The inspector observed the collection and analysis of eddy-current data by contractor personnel and verified that (1) the eddy-current probes being utilized were appropriate for identifying the expected types of indications, (2) probe position location verification was being performed, (3) calibration requirements were being adhered to, and (4) probe travel speed was in accordance with procedural requirements.

The licensee compared flaws detected during the current outage against the previous outage data and that appropriate repair criteria were specified. One hundred percent of all steam generator tubes were inspected during this outage. The inspector noted that the number of tubes required to be plugged was consistent with predictions made prior to the start of the outage. Tube plugging activities during the inspection were in accordance with procedural requirements. Although, still within the allowable limits for tube plugging, the licensee plans to replace both steam generators during the next outage.

The remaining elements of this inspection procedure were addressed during completion of TI 2515/150, “Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles,” and are documented in Section 40A5.3 of this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On June 20, 2005, the inspectors observed licensed operator qualification training activities, including the licensed operators’ performance and the evaluators’ critique (one inspection sample). The inspectors compared performance in the simulator with the Licensed Operator Training Template 84206a, “Station Blackout,” Revision 2, and with performance observed in the control room during this inspection period. The focus of the inspection was on high-risk licensed operator actions and previous lessons-learned items. These items were evaluated to ensure that operator performance was consistent with protection of the reactor core.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

1. Routine Maintenance Effectiveness Inspection

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the requirements of the Maintenance Rule (10 CFR 50.65) and verified that the licensee conducted appropriate evaluations of equipment functional failures, maintenance preventable functional failures, the unplanned capacity loss factor, system unavailability, and classification. The inspectors discussed the evaluations with licensee personnel. The following maintenance rule items were reviewed (two inspection samples):

- Circulating Water Pump CW-1A
- Heatless Air Dryer CA-12

b. Findings

No findings of significance were identified.

2. Biennial Periodic Evaluation Inspection

a. Inspection Scope

The inspectors reviewed the Operating Cycle 21 periodic evaluation that covered the period from September 1, 2002, through December 31, 2003. The inspectors reviewed the program for the monitoring of risk-significant functions associated with structures, systems, and components using reliability and unavailability. The performance monitoring of nonrisk-significant functions using plant level criteria was also reviewed.

The inspectors evaluated whether the report contained adequate assessment of the performance of the Maintenance Rule Program as well as conformance with applicable programmatic and regulatory requirements. To accomplish this, the inspectors verified that the licensee appropriately and correctly addressed the following attributes in the assessment report:

- Program treatment of nonrisk-significant structure, system, and component functions monitored against plant level performance criteria

- Program adjustments made in response to unbalanced reliability and availability

- Application of industry operating experience

- Performance review of Category (a)(1) systems

- Evaluation of the bases for system category status change (e.g., (a)(1) to (a)(2) or (a)(2) to (a)(1))

- Effectiveness of performance and condition monitoring at component, train, system, and plant levels

- Review and adjustment of definitions of functional failures

The inspectors reviewed procedures, condition reports (CRs), and Category (a)(1) recovery plans associated with the above activities for the following

- Air Compressor CA-1B
- Control room air-conditioning units
- Reactor coolant pump seals
- Circulating water pumps
- Safety injection refueling water tank recirculation valves

The inspectors completed 5 of the required 4 to 6 samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed risk assessments by the licensee for equipment outages (four inspection samples) as a result of planned and emergent maintenance to evaluate the licensee’s effectiveness in assessing risk for these activities. The inspectors compared the licensee’s risk assessment and risk management activities against requirements of 10 CFR 50.65 (a)(4). The inspectors discussed the planned and emergent work activities with planning and maintenance personnel. The inspectors verified that plant personnel were aware of the appropriate licensee-established risk category, according to the risk assessment results and licensee program procedures. The inspectors reviewed the effectiveness of risk assessment and risk management for the following activities:

- Replace Component Cooling Heat Exchanger AC-1D CCW Outlet Valve HCV-492B Solenoid, repair CA-7 air compressor for diesel rooms dry pipe sprinkler system, repair FP-181 Fire Hose Cabinet FP-4L hose connection valve, perform Blowdown Tank FW-7 Transfer Pump FW-34B preventive maintenance and main condenser tube leak inspection on June 21, 2005

- Emergent work on nitrogen backup supply to Condensate Makeup Level Control Valve LCV-1109 , preventive maintenance on auxiliary building Supply Air Unit VA-35A motor, and Heated Junction Thermocouple Channels A and B subcooled margin monitors surveillance testing on June 28, 2005
- Routine surveillance testing of Emergency Diesel Generator 2 on June 22, 2005
- Routine maintenance and a full flow test on diesel-driven Auxiliary Feedwater Pump FW-54 on June 30, 2005

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Nonroutine Evolutions and Events (71111.14)

a. Inspection Scope

On March 17, 2005, an event occurred involving the widespread contamination of the auxiliary building and (low level) exposures to several workers. The inspectors reviewed the circumstances involving these events including the licensee's cause determination, the compliance with normal and abnormal operating procedures, and exposures to the individuals.

b. Findings

Introduction. The inspector identified a Green self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, because the licensee's operational procedure for containment building ventilation was inadequate. Specifically the procedure that controlled the containment ventilation fans did not specify the order to start the supply and exhaust fans and resulted in contamination of the auxiliary building.

Description. On March 17, 2005, at approximately 5 a.m., the PING-211 radiation monitor located outside of the main containment hatch in Corridor 26 of the auxiliary building alarmed. A decontamination technician had notified the radiation protection staff that PING-211 was in an alarming condition as early as 4:45 a.m. The radiation protection staff had determined, in error, that the alarm was due to signal noise and reset the radiation monitor. When PING-211 alarmed for a second time, at approximately 5:45 a.m., the radiation protection staff began collecting additional air samples for analysis. At approximately 6 a.m., the licensee announced that all workers were to evacuate both the containment and the auxiliary buildings. Eleven workers received unplanned and unintended low-level intakes (less than 5 millirem) of Co-60. Refer to NRC Inspection Finding 05000285/2005002-06 for a more complete description of the licensee's radiological response to this transient.

The licensee identified the cause of the radioactive airborne condition in the containment and auxiliary buildings to have been ventilation system alignment. On March 17, 2005, at 4:06 a.m. the control room operators had secured the containment building ventilation system. At approximately 5 a.m., when the control room operators restarted the containment building ventilation system they started the supply fans before starting the exhaust fans. Fort Calhoun Operating Instruction OI-VA-1, “Containment Heating, Cooling and Ventilation Systems Normal Operation,” Revision 56, Attachment 8, Step 18 specified “Start one pair of Purge Air Fans: VA-24A and VA-32A, . . .” (VA-24 is the Containment Purge Air Supply Unit and VA-32 is the Containment Purge Air Discharge Unit.) Following the steps exactly as the procedure was written, the operator started the supply fan first. The contamination from the refueling floor and other areas of the containment building became airborne and contaminated the auxiliary building via the personnel access hatch.

Analysis. The inadequate containment building ventilation system operational procedure was a performance deficiency. The finding was more than minor because it affected the Occupational Radiation Safety cornerstone objective to protect worker health and safety from radiation and radioactive materials. Specifically, the Plant Facilities/Equipment Attribute of the cornerstone was affected and involved unplanned and unintended dose to workers. The Occupational Radiation Safety Significance Determination Process was used to analyze the significance of the finding, which was determined to be of very low safety significance because it did not involve: (1) as low as is reasonably achievable (ALARA) planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding had human performance crosscutting aspects in that the subject procedure was inadequate to prevent contaminating the auxiliary building.

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 . . .” Contrary to the above, the licensee’s containment ventilation procedure incorrectly specified the starting sequence of the supply and exhaust fans. The inspector determined that the licensee’s inadequate procedure contributed to 11 workers receiving unplanned and unintended occupational exposure (less than 5 millirem) from airborne Co-60. Because the inadequate procedure resulted in an occurrence of very low safety significance, and it has been entered into the licensee’s corrective action program as CR 200501394, this violation is being treated as a noncited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2005003-01).

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability evaluations (three inspection samples) to verify that the evaluations provided adequate justification that the affected equipment could still meet its Technical Specification, Updated Safety Analysis Report, and design bases requirements. The inspectors also discussed the evaluations with cognizant licensee personnel. The inspectors reviewed the operability evaluations and cause assessments for the following:

- Elevated out board horizontal vibration on Main Feedwater Pump FW-4A (CR 200503223)
- Presence of water in Charging Pump CH-1A lubricating oil sample (CR 200500420)
- Toxic gas protection of the control room ventilation system when inoperability is not due to nonfunctional monitors (CR 200501526)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors performed a review of operator workarounds, control room deficiencies, and control room burden lists. The inspectors focused on the specific effects of a workaround (one inspection sample) on the reliability/availability of mitigating systems and the corresponding impact on operators to respond in a correct and timely manner to plant transients and accidents. The inspector reviewed the effect of frequent voiding conditions on the Low Pressure Safety Injection system against the licensee's Procedure OPD-4-17, "Control Room Deficiencies, Operator Burdens, and Operator Workarounds," Revision 12.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed Work Order 00156552 and the associated engineering change, 10 CFR 50.59 screen and safety evaluation that modified the emergency core cooling system's containment sump screen (one inspection sample). The inspectors performed the review and a walkdown of the installed equipment to ensure that the safety function of the screen was not challenged by the change. The inspectors reviewed the modification against the requirements within 10 CFR 50.59 and discussed the modification with operations and engineering personnel.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Tests (71111.19)

a. Inspection Scope

The inspectors observed and/or reviewed postmaintenance tests (six inspection samples) to verify that the test procedures adequately demonstrated system operability. The inspectors also verified that the tests were adequate for the scope of the maintenance work performed and that the acceptance criteria were clear and consistent with design and licensing basis documents. The following activities were included in the scope of this inspection:

- Work Order 00206032-01, adjust lift on Raw Water Pump AC-10C on May 4, 2005
- Work Order 00205006-01, replace Auto Load Shed Channel A control switch on May 10, 2005
- Work Order 00206035-01, rebuild/replace Containment Spray Header Isolation Valve HCV-344 packing on May 11, 2005
- Work Order 00205692-01, rebuild/replace Containment Spray Header Isolation Valve HCV-345 packing on May 11, 2005
- Work Order 00208560-01, replace reed switch for Control Element Assembly RC-10-41 on June 8, 2005
- Work Order 00209141-01, cut pipe and install caps downstream of Vessel Seal Leakage Instrument Line Waste Drain Line Valve RC-163 on June 13, 2005

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

At the start of the inspection period the plant was in Mode 5 with all fuel off-loaded to the spent fuel pool for the spring 2005 refueling outage. The inspectors observed the core fuel reload, shutdown maintenance activities, reactor heatup, and reactor criticality. Following reactor criticality and operation at power, the reactor was shutdown and cooled down to a cold condition. This allowed replacement of two reactor coolant pump seals on RC-3A and RC-3B while in a risk significant midloop condition. After replacement of the reactor coolant pump seals and other maintenance, the reactor was heated and taken critical.

The inspectors performed several containment tours and verified that activities were performed in accordance with approved procedures and Technical Specification requirements. Periodically, the inspectors evaluated plant conditions to verify that safety systems were properly aligned and that maintenance activities were controlled in accordance with the outage risk control plan.

b. Findings

Introduction. A Green self-revealing noncited violation was identified as a result of the failure of the refueling machine operators to follow the governing procedure, as required by Technical Specification 5.8.1.a. This failure resulted in operators not identifying that Fuel Assembly Y019 was improperly seated in Core Location H17 until the adjacent fuel assembly was loaded and properly seated.

Description. On May 11, 2005, refueling machine operators were reloading fuel assemblies back into the reactor vessel using Procedure OI-FH-1, “Fuel Handling Equipment Operations,” Revision 66. After placing the fourth fuel assembly in the core the operators discovered that Fuel Assembly Y019 was not in the proper position in its core location. The operators had not followed the procedure when placing Fuel Assembly Y019 into Core Location H17. The refueling machine operators failed to compare the Fuel Assembly Y019 cable slack elevation to a value previously recorded when transferring a fuel assembly from the upender to Core Location H17. The procedure required comparing the cable slack elevation to previously recorded values to ensure proper fuel assembly seating on the core support plate. This failure resulted in not identifying that the fuel assembly was improperly seated in Core Location H17. The fuel assembly was being supported on its alignment pins instead of the fuel assembly base. This arrangement could have allowed the assembly to fall over and damage fuel pins.

Analysis. The inspectors determined that the refueling operators failure to follow the fuel handling procedure was a performance deficiency. This finding was considered more than minor because it is associated with the human performance attribute of the barrier integrity cornerstone for fuel cladding. The finding also affects the cornerstone objective of providing reasonable assurance that the fuel cladding will prevent the release of radionuclides caused by accidents or events. The finding was not suitable for analysis under the significance determination process. Regional management review determined that the finding was of very low safety significance (Green) because there was no affect on the reactor coolant system and no radionuclide release occurred.

This finding had crosscutting aspects associated with human performance. The failure of the refueling machine operator’s to follow the procedure for movement of fuel in the reactor vessel directly contributed to the finding.

Enforcement. Technical Specification 5.8.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Appendix A, requires, in part, written procedures for refueling and core alterations. Procedure OI-FH-1, “Fuel Handling Equipment Operations,” Revision 66, in part, requires comparison of the fuel assembly cable slack elevation to previously recorded values when transferring a fuel assembly from the upender to a reactor core location. Contrary to the above, on May 11, 2005, the refueling machine operators failed to compare the fuel assembly Y019 cable slack elevation to a previously recorded value when transferring a fuel assembly from the upender to Core Location H17. This failure resulted in not identifying that the fuel assembly was improperly inserted in Core Location H17 at the time. This violation of Technical Specification 5.8.1.a is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (NCV 285/2005003-02). This violation was entered into the licensee’s corrective action program as CR 200502434.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed and/or reviewed the performance and documentation for the following surveillance tests (three inspection samples) to verify that the structures, systems, and components were capable of performing their intended safety functions and to assess operational readiness:

- Procedure SE-ST-AFW-3009, “AFW Injection Check Valves FW-163 and FW-164 Close Test,” on June 22, 2005
- Procedure OP-ST-RC-3001, “Reactor Coolant System (RCS) Leak Rate Test,” on June 28, 2005
- Procedure IC-ST-AFW-0005, “Channel Check of Auxiliary Feedwater System Flow Transmitters,” on May 2, 2005

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Modification EC 36414 (one inspection sample) that analyzed the use of fuel assembly AA06 with three fuel alignment pins. Normally each fuel assembly has 4 alignment pins that help guide the structure into position on the core support plate when being reloaded into the reactor. During the previous operating cycle one of the alignment pins from fuel assembly AA06 detached and became a loose part within the core. The inspectors reviewed the associated 10 CFR 50.59 evaluation to confirm that the modification (i.e., operating with three versus four pins) had no adverse impact on safety by introducing unanalyzed failure modes.

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 inspector reviewed the Fort Calhoun Station Emergency Plan, Revision 26 to Section B and Revision 11 to Section K, submitted in January 2005, and Revision 16 to Appendix A, submitted in February 2005. Section B was revised to include the field team driver as a fourth field team technician and created an additional operations support center position for dosimetry monitoring and facility accountability. Appendix A was revised to change out letters of agreement which had been renewed.

The revisions were compared to the previous revisions, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, and to the requirements of 10 CFR 50.47(b) to determine if the licensee adequately implemented the emergency plan change process described in 10 CFR 50.54(q).

The inspector completed one sample during the inspection.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

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

- Three outage work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures.
- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling, and engineering groups
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents
- Total Effective Dose Equivalent ALARA evaluations
- Shielding requests and dose/benefit analyses
- 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
- Declared pregnant worker during the current assessment period, monitoring controls, and the exposure results
- Self-assessments and audits related to the ALARA program since the last inspection

The inspector completed 5 of the required 15 samples and 6 of the optional samples.

b. Findings

(1) Introduction. A Green self-revealing noncited violation was reviewed for failure to comply with a radiation work permit requirement. Radiation protection personnel were not consulted or informed of an equipment change for a radiologically controlled activity.

Description. On March 3, 2005, several workers alarmed the personnel contamination monitors upon exiting the radiologically controlled area. The individuals were externally contaminated and four personnel had low levels of internal contamination. The maximum dose assigned was 37 millirem. A subsequent investigation into the event revealed that a job supervisor authorized the change of a cutting tool, from a bandsaw to a grinder, in order to finish cutting out a section of the chemical volume control system piping. Contamination levels were as high as 500 millirad per hour per 100 centimeters square. The gap space for a remaining piece of the piping was too small to finish cutting with a bandsaw and the decision was made to use a grinder to access the remaining piping. However, the job supervisor and the worker did not notify radiation protection of this tool change. The radiation work permit (05-1532, Revision 1) specified that radiation protection was to be contacted prior to starting evolutions which may cause airborne radioactivity. The change from an approved cutting tool, a band saw, to one that had a higher potential for creating airborne radioactivity, a grinder, did not allow radiation protection to assess the radiological protection need for the workers.

Analysis. The failure to follow a radiation work permit requirement is a performance deficiency. This finding was considered more than minor as it was associated with the Occupational Radiation Safety attribute of Program and Process and affected the cornerstone objective to ensure adequate protection of the workers' health and safety from exposure to radiation. The failure to comply with a radiation work permit requirement resulted in the low-level internal contamination of four workers. This event involved workers unplanned, unintended doses that resulted from actions contrary to a radiation work permit requirement which led to the internal contamination of four workers.

This finding was evaluated with the Occupational Radiation Safety Significance Determination Process and was determined to be 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. In addition, this finding had a crosscutting aspect with respect to human performance because the job supervisor or worker did not consult or inform radiation protection before making a change in approved cutting instruments which directly contributed to the finding.

Enforcement. Technical Specification 5.8.1.a states, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, February 1978, Appendix A. Regulatory Guide 1.33, Appendix A, Section 7.e.(1), recommends procedures for a radiation work permit system. Standing Order SO-G-101, "Radiation Worker Practices," Revision 26, Section 5.7.1 states, in part, that radiation work permits are required for entry into any posted radiologically controlled area. Further, Section 5.8.2 of SO-G-101 states, in part, that persons entering a radiologically controlled area shall read and understand the information provided and follow the requirements of the appropriate radiation work permit.

Radiation Work Permit 05-1532, Revision 1, instructed workers to contact radiation protection personnel prior to starting an evolution which may cause airborne radioactivity. The work activity was approved for use with a band saw to minimize the potential of creating a radiological airborne area.

Contrary to this requirement, radiation protection personnel were not contacted before the job supervisor approved a change in cutting tools. Therefore, radiation protection personnel did not have the opportunity to evaluate the possible consequences and implement appropriate protective measures. Consequently, four people received minor uptakes of radioactive material. This finding was placed into the licensee's corrective action program as CR 2005-0943. Because this violation was of very low safety significance and was entered into the licensee's corrective action program, it is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/200503-03, Failure to comply with a radiation work permit requirement.

(2) Introduction. A Green self-revealing noncited violation was reviewed for failure to comply with a Technical Specification required radiation protection procedure. Radiation protection personnel were not informed of an electronic dosimeter dose rate alarm.

Description. On April 7, 2005, a worker received a dose rate alarm and did not notify radiation protection personnel. The worker's electronic dosimeter was found in the dosimeter rack by a shift radiation protection technician active and in a dose rate alarm. When investigated, the worker was identified to have entered a high radiation area to perform his work activity, received a dose rate alarm, and attempted to locate radiation protection near the work area. When unable to do so, the worker left containment and the radiologically controlled area. The individual did not report the alarm to radiation protection personnel during his egress from the radiologically controlled area. During the subsequent investigation, radiation protection personnel were unable to identify a reason for a change in dose rates from those previously surveyed. However, due to the elapsed time between the dose rate alarm and radiation protection personnel becoming aware of the event, changes in radiological conditions for the work area could have occurred without anyone's knowledge. In addition, since the dose rate alarm screen cleared quickly when the individual attempted to log out of the radiologically controlled area, the individual assumed he had successfully logged out of the radiologically controlled area, racked his electronic dosimeter, and left the area.

Analysis. The failure to follow a Technical Specification required radiation work procedure is a performance deficiency. This finding was considered more than minor as it was associated with the Occupational Radiation Safety attribute of Program and Process and affected the cornerstone objective to ensure adequate protection of the worker's health and safety from exposure to radiation. This event involved a worker's unplanned, unintended dose that resulted from actions contrary to radiation work procedures and led to radiation protection personnel investigating the reason for a dose rate alarm in a timely manner.

This finding was evaluated with the Occupational Radiation Safety Significance Determination Process and was determined to be 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. In addition, this finding had a crosscutting aspect with respect to human performance because the worker did not notify radiation protection personnel of a dose rate alarm.

Enforcement. Technical Specification 5.8.1.a states, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, February 1978, Appendix A. Appendix A, Section 7.e.(1), recommends radiation protection procedures for controlling access to radiation areas. Station Procedure SO-G-101, “Radiation Worker Practices,” Revision 26, Section 5.9.2.6, states that if a dose rate alarm is experienced, radiation protection technicians be consulted at the work site or exit the radiologically controlled area and notify the shift radiation protection technician.

Contrary to this requirement, radiation protection personnel were not notified of the dose rate alarm. Therefore, radiation protection personnel did not have the opportunity to timely investigate the radiological conditions which caused the dosimeter to alarm. This finding was placed into the licensee’s corrective action program as CR 2005-1912. Because this violation was of very low safety significance and was entered into the licensee’s corrective action program, it is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000285/200503-04, Failure to comply with a Technical Specification required radiation work procedure.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

1. ALARA Planning and Controls

Section 2OS2 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding exposure tracking, higher than planned exposure levels, and radiation worker practices. The inspector reviewed the corrective action documents listed in the attachment against the licensee’s problem identification and resolution program requirements. No findings of significance were identified.

2. Resident Inspector Selected Issue Follow-up

The inspectors selected one issue (one inspection sample) for a more in-depth review to verify that the licensee personnel had taken corrective actions commensurate with the significance of the issue. On March 25, 2004, control room Air Conditioning Unit VA-46B tripped during the performance of preventive maintenance

(CR 200401148). The inspectors reviewed the corrective actions associated with this CR. When evaluating the effectiveness of the licensee's corrective actions, the following attributes were considered:

- Complete and accurate identification of the problem in a timely manner commensurate with its significance and ease of discovery
- Evaluation and disposition of operability and reportability issues
- Consideration of extent of condition, generic implications, common cause, and previous occurrences
- Classification and prioritization of the resolution of the problem commensurate with its safety significance
- Identification of corrective actions which are appropriately focused to correct the problem
- Completion of corrective actions in a timely manner commensurate with the safety significance of the issue

3. Routine Review of Maintenance Rule Identification and Resolution of Problems

As part of the Maintenance Rule biennial periodic evaluation inspection (Section 1R12), the inspectors evaluated the use of the corrective action system within the Maintenance Rule program for issues associated with risk significant systems. The review was accomplished by the examination of a sample of corrective action documents associated with systems which are or had been in Maintenance Rule Category (a)(1), including recovery plans for improving system performance. The purpose of the review was to establish that the corrective action program was entered at the appropriate threshold for the purpose of:

- Implementation of the corrective action process when a performance criterion was exceeded
- Correction of performance-related issues or conditions identified during the periodic evaluation
- Correction of generic issues or conditions identified during programmatic assessments, audits, or surveillances

The inspectors reviewed the following documents to evaluate implementation of the corrective action process. Specifically, the inspectors selected 20 CRs/cause determinations from Expert Technical Panel meeting minutes; 8 CRs related to operability evaluations; 14 CRs from a list of the raw water, high pressure safety injection, and 125 Vdc systems; 10 CRs from the Maintenance Rule coordinator's database; and selected CRs from the quality assurance audits and from the last completed periodic assessment.

4. Inservice Inspection Activities

The inspector reviewed inservice inspection related CRs issued during the current and past refueling outages and verified that the licensee identified, evaluated, corrected, and trended problems. The review included an evaluation of the effectiveness of the licensee's corrective action process, including the adequacy of the technical resolutions.

5. Semiannual Trend Review

The inspectors performed a semiannual assessment (one inspection sample) of the licensee's corrective action program. The assessment covered open CRs written since the 2003 refueling outage to determine if appropriate prioritization and timely corrective actions were scheduled to correct outstanding conditions. The focus of the inspection was on conditions considered by the licensee as conditions adverse to quality where immediate corrective actions had been completed and documented but did not address the condition.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

(Closed) Licensee Event Report 05000285/2004001-00, Failure to Perform a Leakage Test Due to Lack of Understanding of Penetration Design

While investigating the requirements for the Type B Local Leak Rate Test for the fuel transfer tube between containment and the auxiliary building, the licensee discovered that the sleeve for the tube had not been properly tested since initial construction of the plant. (Type B tests are normally performed on a once per refueling cycle periodicity.) The licensee determined the cause to be a lack of knowledge or understanding of the design features of the penetration. The fuel transfer sleeve was subsequently tested satisfactorily. The licensee reviewed similar penetrations to ensure that this problem had not occurred at any other locations. No similar situations were identified. This finding is more than minor because it had a credible impact on safety, in that if an accident occurred, containment integrity could not have been assured. This finding affects the Barrier Integrity Cornerstone and was considered to have very low safety significance (Green) using Appendix H of the significance determination process. This conclusion was based on the containment penetration having been subsequently determined to be functional and was a barrier to the auxiliary building versus the environment. This licensee-identified finding involved a violation of Technical

Specification 3.5 (3) (iv), “Containment Test,” and 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.” The enforcement aspects of the violation are discussed in Section 4OA7 of this report. This licensee event report is closed.

4OA4 Crosscutting Aspects of Findings

Section 1R14 describes the consequences associated with an inadequate operating procedure for the containment building ventilation system. This resulted in contamination of the auxiliary building.

Section 1R20 describes the failure of the refueling machine operators to follow the procedure for transferring fuel in the reactor vessel as required by Technical Specification 5.8.1.a. This human performance failure resulted in not identifying that Fuel Assembly Y019 was improperly seated into Core Location H17.

Section 2OS2 describes two occupational radiation safety findings. The first finding was associated with a failure to comply with a radiation work permit, and the second involved a failure to comply with a Technical Specification required radiation work procedure. Both of these findings had human performance crosscutting aspects to them.

Section 4OA5 describes the failure to verify that outside contractors were properly qualified to perform safety significant activities and the failure to certify nondestructive testing personnel in accordance with ASME requirements.

4OA5 Other Activities

1. Temporary Instruction 2515/161 - Transportation of Reactor Control Rod Drives in Type A Packages

a. Inspection Scope

This area was inspected to verify that the licensee’s radioactive material transportation program complies with specific requirements of 10 CFR Parts 20, 71, and Department of Transportation regulations contained in 49 CFR Part 173. The inspector interviewed licensee personnel to determine if the licensee had undergone refueling/defueling activities between January 1, 2002, and present, and whether they had shipped irradiated control rod drives in Department of Transportation Specification 7A Type A packages.

b. Findings

No findings of significance were identified.

2. Temporary Instruction 2515/163 - Operational Readiness of Offsite Power

a. Inspection Scope

The inspectors collected data pursuant to Temporary Instruction 2515/163, "Operational Readiness of Offsite Power." The inspectors reviewed the licensee's procedures related to General Design Criteria 17, "Electric Power Systems;" 10 CFR 50.63, "Loss of All Alternating Current Power;" 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants;" and the Technical Specifications for the offsite power system. The data was provided to the Office of Nuclear Reactor Regulation for further review. Documents reviewed for this Temporary Instruction are listed in the attachment.

b. Findings

No findings of significance were identified.

3. Reactor Pressure Vessel Head and Vessel Head Penetration (VHP) Nozzles (Temporary Instruction 2515/150, Revision 3)

a. Inspection Scope

The inspector reviewed the licensee's reactor VHP nozzle inspection activities implemented in accordance with the requirements of NRC Order EA-03-009, issued on February 20, 2004. The NRC's follow-up of the licensee's activities are delineated in Temporary Instruction 2515/150, Revision 3, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles." The licensee performed a visual and nonvisual inspection.

The visual inspection was a bare metal visual examination that verified the absence of boron crystals and the integrity of the reactor pressure vessel head. The inspectors independently observed approximately 80 percent of the bare metal visual examination and five vessel head nozzle examinations.

The licensee used a robotic device and a borescope to perform the vessel head inspection. The robotic device performed a 360 degree inspection around each nozzle penetration that it could access. The penetrations near the reactor pressure vessel head outer edge, where the stepped reflective insulation met the vessel head, could not be inspected by the robotic device alone due to clearances and the slope of the head. These areas were inspected using a borescope that was attached to the robotic device. This combination allowed accurate identification of the penetration to be inspected and surrounding area. It also, provided additional light for the borescope and stabilized the borescope. The use of the borescope attached to the robotic device was a lesson learned from previous inspections that was incorporated in this inspection.

The licensee's quality control personnel involved with the bare metal visual examination inspection were VT-2 qualified and performed the previous bare metal visual inspection. They were familiar with the robotic setup and the limitations of the robot through previous training and experience.

The licensee could detect small boron deposits as described in NRC Bulletin 2001-001. No evidence of boric acid deposits were found. However, the licensee observed boric acid stains in some locations on the reactor pressure vessel head and on some nozzles that were associated with past flange leakage from above. The vessel head contained small debris, dust of light crystals, and a few foreign objects such as small mechanical fasteners. The debris and light crystals were easily scattered with air. No deficiencies were identified that required repairs.

The non-visual nondestructive examination technique was a surface examination using eddy-current testing of the wetted surface of the VHP nozzle base material and the J-groove weld.

The temporary instruction procedure requires review of 10 percent of vessel head nozzle volumetric and 5-10 percent of nozzle and/or J-groove surface examinations. The inspector reviewed surface examination results for 9 of 47 nozzles (19 percent) and 5 of 47 (10.6 percent) J-groove welds. Four indications of shallow, circumferentially oriented surface scratches at the outside edge of the J-groove weld in the weld cover had been identified. The licensee planned to retest these areas and further evaluate the significance of the indications. The review and acceptance of the eddy-current data results had not been completed at the end of this inspection. Consequently, the evaluation of all information and entry of appropriate conditions into the corrective action system for the affected nozzles had not occurred.

The temporary instruction procedure requires review of one or two examinations from the previous outage with recordable indications from surface and volumetric examinations, if applicable. There were no volumetric or surface examinations with recordable indications from the previous outage.

The temporary instruction procedure requires an independent review of the licensee's implementation of the chosen method to detect relevant surface conditions. As part of this review the inspector reviewed the contractor's personnel certifications, procedures and calibrations.

b. Findings

During the review of the licensee's reactor VHP nozzle inspection activities, the following findings were identified.

1. Introduction. The inspector identified a noncited violation of very low safety significance (Green) for failure to ensure that an outside contractor was properly qualified to perform

safety significant activities under the licensee's established quality assurance plan. This failure violated Standing Order SO-G-72, "Special Process Control."

Description. During a review of records associated with the surface examination of the wetted surface of the VHP nozzles and associated J-groove welds, the inspector determined that the licensee had not verified that personnel certifications, equipment calibrations and procedures of the company contracted to perform the inspection, R. Brooks Associates, Inc., met the requirements prior to performing work, as required by Standing Order SO-G-72.

R. Brooks Associates, Inc. was contracted by Omaha Public Power District to perform eddy-current testing of the VHP nozzles and associated J-groove welds. However, because R. Brooks Associates, Inc., did not have a quality assurance plan certified as conforming to 10 CFR Part 50, Appendix B, they were not qualified to perform safety-related work under their own quality assurance plan. Instead, Omaha Public Power District required that R. Brooks conform to their quality assurance program requirements, as allowed by Standing Order SO-G-72, and the quality assurance plan.

Section 5.1.2 of Standing Order SO-G-72 requires that, if an outside organization is contracted to perform a special process, such as eddy-current testing, the responsible group (in this case the engineering department) ensures that the quality control department has verified that the contractor's personnel certifications, equipment calibrations, and procedures meet requirements prior to performing work. This is intended to ensure that personnel certifications, equipment calibrations and procedures conform to the quality assurance plan.

Contrary to Section 5.1.2 of SO-G-72, the quality control department did not review and approve the R. Brooks' personnel certifications, equipment calibrations and procedures prior to performing work. This included the verification and validation of the computer software utilized by R. Brooks Associates, Inc.

Analysis. The failure to follow the requirements of Standing Order SO-G-72 to review and approve the R. Brooks personnel certifications, equipment calibrations, and procedures was a performance deficiency. This finding is greater than minor because it affected an attribute and the objective of the barrier integrity cornerstone. Specifically, inspections of reactor pressure vessel penetrations were performed by a subcontractor who may or may not be technically qualified to perform such inspections and, as a result, verification that reactor coolant system leakage from VHP nozzles would not occur, was not assured. The finding has very low safety significance because the plant was in an outage and the licensee entered the finding into their corrective action program for disposition prior to restart.

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to this, the licensee failed to follow procedures requiring verification of the qualifications of R. Brooks Associates, Inc. personnel, calibrations and procedures prior to starting work on safety-related systems. Because the violation was of very low safety significance and has been entered into the licensee's corrective actions program this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/2005003-005.

2. Introduction. The inspector identified a noncited violation of very low safety significance (Green) for failure to certify nondestructive testing personnel in accordance with the American Society of Nondestructive Testing (ASNT), "Standard for Qualification and Certification of Nondestructive Testing Personnel," CP-189-1991. The requirement for certification under this industry standard is included in Quality Assurance Plan QAP-9.3, "Training and Certification of Inspectors."

   Description. During a review of records associated with the eddy-current testing surface examination of the wetted surface of the reactor VHP nozzles and associated J-groove welds, the inspector determined that the qualification of R. Brooks Associates, Inc., Level 2 nondestructive testing personnel was not in accordance with ASNT CP-189-1991 requirements.

   R. Brooks was contracted by Omaha Public Power District to perform eddy-current testing of the VHP nozzles and J-groove welds under the umbrella of the quality assurance program. The program requires that nondestructive testing personnel meet the requirements of ASNT CP-189-1991.

   Section 6.3.3 of CP-189-1991 outlines the requirements for practical examinations of Level 2 nondestructive testing personnel. This section requires that "the candidate shall demonstrate proficiency by performing the applicable nondestructive test method in examining at least one sample per technique and a minimum of two samples per method . . . The test samples shall be representative of the product that the candidate will encounter in performing the job functions."

   The reviewed records indicated that the two Level 2 nondestructive testing personnel employed by R. Brooks were each examined with only one sample. This failed to satisfy the ASNT CP-189-1991 requirements of a minimum of two samples per method. In addition, one employee was tested using a calibration block sample, and the other was examined with an outside diameter calibration specimen. Neither of these samples was fully representative of the products encountered during the inside diameter, outside diameter and J-groove weld examinations.

   Analysis. The inspector determined that failure to ensure the certifications of Level 2 nondestructive testing personnel was a performance deficiency. This finding is greater than minor because it affected an attribute and the objective of the barrier integrity cornerstone. Specifically, inspections of wetted surface of the reactor VHP nozzles and associated J-groove welds were performed by a personnel who may or may not be technically qualified to perform such examinations, and as a result, verification that reactor coolant system leakage from VHP nozzles would not occur was not assured. The finding has very low safety significance because the plant was in an outage and the licensee entered the finding into their corrective action program for disposition prior to restart.

Enforcement. 10 CFR Part 50, Appendix B, Criterion IX, states, that "measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Contrary to this, the licensee failed to satisfy the requirements of ASNT CP-189-1991, for a practical examination to consist of a minimum of two samples representative of the products a candidate will encounter when performing a job. Because the violation was of very low safety significance and has been entered into the licensee's corrective actions program this violation is being treated as an noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000285/2005003-06.

4OA6 Meetings

Exit Meeting Summaries

On April 8, 2005, the health physicist inspector presented the ALARA inspection results to Mr. D. Bannister, Plant Manager, and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

The reactor inspector presented the results of the inservice inspection effort to Mr. D. Bannister, Plant Manager, and other members of licensee management on April 8, 2005. The inspector asked the licensee whether any material examined during the inspection should be considered proprietary. Several documents were identified as containing proprietary information by the licensee. The inspector informed the licensee that these documents would be destroyed upon completion of their review.

On May 10, 2005, the health physics inspector discussed the inspection findings with Mr. M. Puckett, Radiation Protection Manager. The inspector verified that no proprietary information was provided during the inspection.

On May 12, 2005, the emergency preparedness inspector discussed the inspection findings with Mr. C. Simmons, Supervisor, Emergency Planning. The inspector verified that no proprietary information was provided during the inspection.

The reactor inspectors presented the inspection results to Mr. D. Banister, Plant Manager, and other members of licensee management at the conclusion of the Maintenance Rule biennial periodic evaluation inspection on June 17, 2005. No proprietary information was reviewed.

The resident inspectors presented the results of the quarterly inspection effort to Mr. D. Bannister, Plant Manager, and other members of licensee management on July 6, 2005. The inspectors confirmed that proprietary information examined during the inspection had been returned to the licensee. Licensee management acknowledged the inspection findings.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- Technical Specification 3.5 (3) (iv), “Containment Testing,” required that visual examinations and leakage rate testing be conducted on listed penetrations. Contrary to the above, the licensee failed to test the fuel transfer tube (Mechanical Penetration M - 100) since initial construction of the plant. This finding only had very low safety significance because the containment penetration was determined to be functional and was a barrier to the auxiliary building versus the environment. This finding was identified in the licensee’s corrective action program as CR 200402619 and was reported as Licensee Event Report 50-285/2004-001-00.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bannister, Plant Manager
B. Blome, Manager, Planning
G. Cavanaugh, Supervisor, Station Licensing
A. Clark, Manager, Security and Emergency Planning
M Core, Manager, System Engineering
P. Cronin, Manager, Shift Operations
D. Dryden, Licensing Engineer
H. Faulhaber, Manager, Work Management
M. Frans, Assistant Plant Manager
W. Goodell, Manager, Operations
P. Hamer, System Engineer, ISI Component Testing
R. Haug, Manager, Chemistry
R. Hawkins, Senior QA Lead Auditor
J. Herman, Manager, Nuclear Licensing
J. Kellams, Acting Manager, Corrective Action Group
J. Mathew, System Engineer, Steam Generators
T. Matthews, Supervisor, Nuclear Licensing
E. Matzke, Station Licensing Engineer
J. McManis, Manager, Design Engineering
G. Miller, ISI Coordinator
R. Perry, IC Supervisor
R. Phelps, Division Manager, Nuclear Engineering
T. Pilmaier, Manager, Corrective Action Group
M. Puckett, Manager, Radiation Protection
R. Ruhge, Supervisor, Quality Control
C. Simmons, Supervisor, Emergency Planning
J. Skiles, Manager, Design Engineering
S. Sterba, Supervisor, Design Engineering
S. Swearngin, Supervisor, Reliability Engineering
R. Tella, Engineer, Reliability Engineering
M. Tesar, Division Manager, Nuclear Support
J. W. Tillis, Manager, Maintenance
D. Trausch, Manager, Quality Assurance
P. Turner, System Engineer
R. Westcott, Manager, Training
K. Woods, Design Engineer, Reactor Vessel Head Inspection
J. Zagata, Engineer, Reliability Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000285/2005003-01	NCV	Inadequate ventilation procedure resulting in internal contaminations to personnel (Section 1R14)
05000285/2005003-02	NCV	Failure to follow the procedure for transferring fuel in the reactor vessel (Section 1R20)
05000285/2005003-03	NCV	Failure to comply with a radiation work permit requirement (Section 2OS2)
05000285/2005003-04	NCV	Failure to comply with a Technical Specification required radiation work procedure (Section 2OS2)
05000285/2005003-05	NCV	Failure to follow procedures for ensuring qualification of contractors (Section 4OA5)
05000285/2005003-06	NCV	Failure to certify nondestructive testing personnel (Section 4OA5)

Closed

05000285/2004001-00	LER	Failure to perform a leakage test due to lack of understanding of penetration design

LIST OF DOCUMENTS REVIEWED

Resident Inspector Baseline Inspections

Licensee Procedure SO-G-28, “Fort Calhoun Station Standing Order,” Revision 61

Conduit Installation Data Sheet for Conduit Identifications EB3545, EB3557, and EB11428 dated March 22, 2005

Drawing Number 16047, “Auxiliary Boiler Fuel Oil Tank,” Revision 2

Drawing Number 13229, “Diesel Fuel Oil Tank FO-1,” Revision 3

Section 1R08 Inservice Inspection Activities (71111.08)

Procedures (Omaha Public Power District)	Title	Revision
SO-G-72	Special Process Control	18
QAP-1.3	Quality Assurance Program Boundary	5
QAP-2.1	Procedures and Instructions	6
QAP-4.2	Approval and Control of Vendors	8
QAP-6.6	Control of Special Processes	4
QAP-8.1	Quality Control Inspection	4
QAP-9.3	Training and Certification of Inspectors	8
05-QUA-015	Quality Refueling Outage Oversight Plan	2/21/05
PED-QP-6	Procurement Requirements-Materials and Services	15
NPM-300	Procurement of Materials and Services	8
NPM-401	Approval of Suppliers	8

Procedures R. Brooks Associates, Inc.	Title	Revision
83-0041	Reactor Vessel Head Remote Visual Inspection	2
83-0070	Remote Eddy Current Examination for Reactor Vessel Head Penetration Nozzle Wetted Surface	1
83-0071	Eddy Current Analysis For Reactor Vessel Head Penetration Nozzle Wetted Surface	0

Procedures (Integrated Technologies, Inc.)	Title	Revision
ET001	Procedure For Analog And Digital Eddy Current Inspection Of Heat Exchanger Tubes	1

Procedures (Washington Group International)	Title	Revision
OPPD-MT-98-1	Magnetic Particle Examination Of Welds And Bolting	1
OPPD-PT-98-1	Liquid Penetrant Examination - Solvent Removable, Visible Dye Technique	0
OPPD-UT-98-1	Manual Ultrasonic Examination Of Ferritic Piping Welds	0
OPPD-UT-98-2	Manual Ultrasonic Examination Of Austenitic Piping Welds	0
OPPD-UT-98-5	Ultrasonic Examination Of Studs/Bolts Greater Than Two Inches In Diameter	0
OPPD-UT-98-9	Ultrasonic Examination Of Cast Austenitic Piping Welds And Vessel Welds Equal To Or Less Than 2 Inches	0
OPPD-UT-98-11	Ultrasonic Straight Beam Examination	0
OPPD-UT-98-12	Ultrasonic Examination Of Class 1 & 2 Vessel Welds Over 2 Inches Thick	0
OPPD-UT-98-14	Manual Ultrasonic Examination Of Nozzle Inside Radius	0

Condition Reports

200303438	200304915	200400214	200401429	200501035
200304316	200305039	200400318	200402323	200501121
200304388	200305051	200400949	200402589	200501544
200304391	200305281	200401042	200402619	200501805
200304503	200305443	200401146	200403597	200501933
200304520	200305704	200401155	200500547	200501957
200304804				

ASME Weld Repair/Replacement Documentation Form

04-7-001
04-7-012
04-7-013
04-7-022
04-7-029

Calibration Data Sheet

SG-1-4B, 5-31
SG-1-C-2, 5-30

Section 2OS2: ALARA Planning and Controls (71121.02)

ALARA Packages

05-12 Reactor Head Removal, Replacement, and Associated Tasks
05-42 Cleaning Activities Under the Reactor Head

Corrective Action Documents

2004-2812, 2004-3476, 2004-3684, 2004-3714, 2004-3902, 2005-0139, 2005-0624, 2005-0507, 2005-0802, 2005-0844, 2005-0943, 2005-0964, 2005-1288, 2005-1330, 2005-1350, 2005-1368, 2005-1479, 2005-1912

Procedures

RP-201	Radiation Work Permits, Revision 27
RP-307	Use and Control of Temporary Lead Shielding, Revision 11
RP-600	Dosimetry Program, Revision 17
RP-602	Personnel Dosimetry Issuance and Change Out, Revision 19
RP-608	Dose Calculations from Contamination, Revision 11
RP-650	Internal Dosimetry Program, Revision 9
SO-G-101	Radiation Worker Practices, Revision 26

Quality Observations and Self-Assessments

2004 Quality Surveillance Observations 556 and 574
2005 Fort Calhoun ALARA and Radiation Protection Program Goals and Implementation Tools
2005 Quality Surveillance Observations 12, 21, and 40
SA-04-050 2004 Self Assessment - Radiation Protection Program

Radiation Work Permits

05-2512	Reactor Head Removal and Replacement Tasks
05-3512	Reactor Head Removal and Replacement Tasks in Restricted High Radiation Areas
05-3534	Under Vessel Inspections and Associated Tasks

Temporary Shielding Requests

05-05, 05-06, 05-12, 05-35, 05-56, and 05-66

Miscellaneous

ALARA Committee Meeting Minutes dated May 11, 2004, through March 30, 2005
Exposure records for one Declared Pregnant Female

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
CFR	*Code of Federal Regulations*
CR	condition report
NCV	noncited violation
NRC	Nuclear Regulatory Commission
VHP	vessel head penetration