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

May 12, 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 05000285/2005002

Dear Mr. Ridenoure:

On March 31, 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 April 11, 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 six NRC-identified findings that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that six violations are associated with these issues. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. 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, DC 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, DC 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/

Michael C. Hay, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2005002
w/attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285
License:	DPR-40
Report:	05000285/2005002
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:	January 1 through March 31, 2005
Inspectors:	J. Hanna, Senior Resident Inspector L. Willoughby, Resident Inspector W. Sifre, Reactor Inspector L. Carson, Senior Health Physicist
Approved By:	Michael C. Hay, Chief, Project Branch C Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000285/2005002; 01/01/2005 - 03/31/2005; 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 an announced inspection by a regional health physicist inspector and reactor inspector. Six Green noncited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A noncited violation of Technical Specification 5.8.1.c, Fire Protection Program Implementation, was identified for the failure follow the fire protection program after exceeding the transient combustibles limit in Room 59. The licensee did not evaluate and establish compensatory measures prior to storing transient combustibles in Room 59 as required by Procedure SO-G-91, “Control and Transportation of Combustible Materials,” Revision 20.

  This finding was more than minor since it was associated with the protection against external factors attribute of the mitigating systems cornerstone. Using the Significance Determination Process, Manual Chapter 0609, Appendix F, the finding was determined to be in the Fire Prevention and Administrative Controls category because it affected the administrative controls used in fire prevention. The degradation rating of the finding was low. This was based on the materials being stored in a room with no heat source and the materials did not contain combustible liquids or were not self heating. The finding was characterized under the significance determination process as having very low safety significance (Green) since the degradation rating was low. Based on previous opportunities for personnel to recognize this condition, a human performance aspect was identified for this finding. This condition has been entered into the licensee’s corrective action program (Section 1R05.b2).

Cornerstone: Mitigating Systems

- Green. A noncited violation of Technical Specification 5.8.1.c, Fire Protection Program Implementation, was identified for the failure to implement procedures to ensure that fire barriers protecting safety-related areas were functional. Specifically, between Rooms 1 and 58, and between Rooms 1 and 30, openings existed in a barrier that would have allowed flame propagation between two respective fire areas.

This finding was more than minor since it was associated with the protection against external factors attribute of the mitigating systems cornerstone. Since the finding occurred while shutdown, Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process, is not applicable for determining the significance of the issue. Regional management determined that the finding was of very low significance (Green). The finding was evaluated considering Manual Chapter 0609, Appendix F as a bounding case and was used as guidance to determine the significance of the finding. The finding was determined to be in the fire confinement category because the fire barrier separated one fire area from another. The inspectors assigned a moderate degradation rating since there was defense-in-depth and no potential damage targets in the exposed fire area that were unique from those in the exposing fire area. The inspectors, using a deterministic process and the guidance of the Phase 1 qualitative screening check, characterized the finding as having very low safety significance (Green) since the distance between safety-related components would protect the equipment in the exposed fire area. This condition has been entered into the licensee's corrective action program (Section 1R05.b1).

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion III, was identified based on the licensee's failure to translate design basis information into specification drawings, procedures, and instructions. Specifically, the licensee failed to maintain design control of the turbine-driven auxiliary feedwater pump to ensure turbine casing condensate drains would function during accident conditions involving loss of condenser vacuum.

  The performance deficiency was a failure to translate the design basis of the plant to maintain the function of the auxiliary feedwater system during a loss of offsite power or other event causing a loss of condenser vacuum. This finding was more than minor because it was similar to Example 3.a of Appendix E in Inspection Manual Chapter 0612. The issue screened out as a Green finding because it was a design or qualification deficiency that was confirmed not to result in a loss of function as defined by NRC Generic Letter 91-18. Based on previous opportunities to recognize and correct this condition, a problem identification and resolution aspect was identified for this finding. This condition has been entered into the licensee's corrective action program (Section 1R15.1).

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion V, was identified based on the licensee's procedures not including appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the containment protective coatings procedure did not contain appropriate criteria to inspect the condition of safety-related coatings.

  This finding affected the Mitigating Systems cornerstone and was considered more than minor because it affected the Procedure Quality attribute of the cornerstone. Specifically appropriate quantitative acceptance criteria was not provided to ensure that representative areas were selected for review within the coatings program. The finding was characterized under the significance determination process as having very low safety significance because the as-found reactor vessel head paint condition did not challenge the debris loading assumptions of the containment sumps and no actual loss of safety function occurred. Based on previous opportunities to recognize and correct this condition, a problem identification and resolution aspect was identified for this finding. This condition has been entered into the licensee's corrective action program (Section 1R15.2).

Cornerstone: Occupational Radiation Safety

- Green. A self-revealing, noncited violation was reviewed because the licensee failed to conspicuously post, barricade, lock or guard a restricted high radiation area per Technical Specifications 5.11.1 and 5.11.2. On March 4, 2005, a worker unexpectedly received an electronic dosimeter dose rate alarm when he entered the lower elevation of the Steam Generator A bay area. Subsequently, the licensee found dose rates that measured 1,500 to 2,000 millirem per hour at 30 centimeters in the area of Valve RC-163 and posted and barricaded the area.

  This finding is more than minor because it affected the Occupational Radiation Safety cornerstone objective to protect worker health and safety from radiation and radioactive materials. This finding was associated with the cornerstone attribute of Exposure Control and involved unplanned and unintended dose to a worker. 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) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding also had crosscutting aspects associated with human performance. The radiation protection organization did not inform its technicians about changing radiological conditions in the area of Valve RC-163 due to plant operations and based on historical data. This occurrence was entered into the licensee's corrective action program (Section 2OS1.1).

- Green. An NRC-Identified, noncited violation of 10 CFR 20.1501(a) was identified because the licensee's radiation protection staff failed to perform an adequate survey to evaluate radiological hazards. Specifically, on March 17, 2005, at approximately 5 a.m. the particulate, iodine, and noble-gas radiation monitor located outside of the main containment hatch alarmed. The radiation monitor indicated increasing airborne radioactivity starting at 3:30 a.m.; however, the licensee did not evaluate the cause of the alarm until 6 a.m. Consequently, 11 workers received unplanned and unintended low-level intakes (less than 5 millirem) of Co-60 because the extent of potential radiological hazards was not fully evaluated.

This finding is more than minor because it affected the Occupational Radiation Safety cornerstone objective to protect worker health and safety from radiation and radioactive materials. This finding was associated with the cornerstone attribute of exposure control 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) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding also had crosscutting aspects associated with human performance. The radiation protection organization did not have an effective process for its technicians to evaluate potential radiological hazards associated with alarming airborne radiation monitors. This occurrence was entered into the licensee's corrective action program (Section 2OS1.2).

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The plant operated at full power until February 25, 2005, when reactor power was decreased in anticipation of the spring 2005 outage. On February 26, 2005, the reactor automatically tripped during the shutdown due to improper operation of a main feedwater regulating bypass valve. At the close of the inspection period, the plant was in Mode 5 with all fuel off-loaded to the spent fuel pool.

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 cold weather protection features (one inspection sample) in response to substantial snow and cold weather on January 6, 2005. The inspector performed a walkdown of accessible areas with the cognizant engineer to verify that cold weather preparations were performed. The inspector reviewed the completed Attachment 1 to Procedure OI-EW-1, "Cold Weather Preparation," Revision 11. The inspector verified that heat tracing and heaters were monitored and verified in accordance with the procedure.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Equipment Walkdowns

a. Inspection Scope

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:

- Low Pressure Safety Injection Train A while Low Pressure Safety Injection Train B was providing shutdown cooling to the core

- Temporary spent fuel pool cooling system while the raw water and component cooling water systems were secured for maintenance

- Boric acid addition paths while Low Pressure Safety Injection Train B was providing shutdown cooling to the core

b. Findings

No findings of significance were identified.

.2 Complete System Walkdowns

a. Inspection Scope

The inspectors conducted a detailed review of the alignment and condition of the emergency diesel generator system (one inspection sample). The inspectors reviewed open work orders and condition reports associated with the system. The inspectors performed a walkdown of accessible portions of the system. During the walkdown, inspectors verified that the system was properly aligned in accordance with piping and instrumentation drawings and operational procedures. Refer to Supplemental Information at the end of this inspection report for a complete list of documents reviewed.

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 34.B1 - Upper Electrical Penetration Room (Room 57)
- Fire Area 36A - East Switch Gear Room (Room 56)
- Fire Area 30 - Containment (Room 1)
- Fire Area 6.7 - Letdown Heat Exchanger Room (Room 12)
- Fire Area 20.3 - Volume Control Tank Room (Room 29)
- Fire Area 23 - Pipe Penetration Room (Room 59)

b. Findings

Two findings of significance were identified.

(1) Fire Barrier Separating Rooms 1 and 58

Introduction. A Green, noncited violation (NCV) of Technical Specification 5.8.1 was identified for the failure to ensure that all fire barriers protecting safety-related areas were functional. Specifically, between Rooms 1 and 58, an opening existed in a barrier due to the personnel access hatch door being open. This condition would have allowed flame propagation between Fire Area 30 (Room 1 - Containment) and Fire Area 20.1 (Room 58 - Auxiliary Building East Corridor 26 and Personnel Access Lock Area, Elevation 1007').

Description. Fire Barrier AE2 is a rated fire barrier that separates Rooms 1 and 58. This barrier is a personnel hatch for ingress and egress into Room 1 (the containment building) and is constructed of two watertight doors connected with an interlocking mechanism. The hatch was open, when the inspectors discovered the condition, to allow personnel and some equipment movement during the licensee's spring 2005 refueling outage. The inspectors inquired with the licensee's fire protection engineer as to whether a fire impairment existed for this fire barrier breach. The fire protection engineer subsequently determined there was none and subsequently initiated an impairment and a condition report documenting the adverse condition.

The inspectors also identified to the fire protection engineer another impairment that also occurred on March 8, 2005. The licensee had the fuel transfer tube flange removed and Valve FH-11 open with no water in the transfer canal. This condition was another breach of the fire barrier between the containment building and the auxiliary building (Room 3). This condition is being considered another example of one fire protection finding/violation. The inspectors also noted, based on a review of the licensee's corrective action program, that several other fire protection impairments had not been established as required during the refueling outage. Further, the licensee had identified an adverse trend associated with fire barrier penetrations, fire doors, fire protection or suppression equipment, etc., as documented in Condition Report 200500222.

Analysis. The inspectors evaluated the safety significance of the finding. This finding affected the mitigating systems cornerstone and was considered more than minor since it affected the cornerstone attribute of protection against external factors. Since the finding occurred while shutdown, Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," is not applicable for determining the significance of the issue. Regional management determined that the finding was of very low significance (Green). The finding was evaluated considering Manual Chapter 0609, Appendix F, as a bounding case and was used as guidance to determine the significance of the finding. The finding was determined to be in the fire confinement category because the fire barrier separated one fire area from another. The inspectors assigned a moderate degradation rating since there was defense-in-depth and no potential damage targets in the exposed fire area that were unique from those in the exposing fire area. The inspectors using a deterministic process and the guidance of the Phase 1 qualitative screening check characterized the finding as having very low safety significance (Green) since the distance between safety-related components would protect the equipment in the exposed fire area.

Enforcement. Technical Specification 5.8.1.c requires, in part, that written procedures shall be established and maintained for implementation of the fire protection program. Procedure SO-G-102, "Fire Protection Program Plan," Revision 6, was the governing document for all fire protection program plan implementing procedures and references Procedure SO-G-103, "Fire Protection Operability Criteria and Surveillance Requirements," Revision 17, which implements fire protection requirements. Procedure SO-G-103, Attachment 7.5, requires, in part, that all fire barriers protecting safety-related areas shall be functional. Contrary to the above, the licensee failed to ensure that all fire barriers protecting safety-related areas were functional. Specifically, between Rooms 1 and 58, and between Rooms 1 and 3, openings existed in fire barriers that would have allowed flame propagation. This violation of Technical Specification 5.8.1 is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2005002-01). This violation is in the licensee's corrective action program as Condition Reports 200501068 and 200501069.

(2) Exceeding Transient Combustibles Limits

Introduction. A Green NCV of Technical Specification 5.8.1 was identified for the failure to follow the fire protection program after exceeding the transient combustibles limit in Room 59. The licensee did not evaluate and establish compensatory measures prior to storing transient combustibles in Room 59 as required by Procedure SO-G-91, "Control and Transportation of Combustible Materials," Revision 20.

Description. On February 18, 2005, the inspectors identified that the temporary storage of the steam generator nozzle dams containing rubber diaphragms and hoses exceeded the 100 pound limit for transient combustibles in Room 59. The inspectors determined that no fire impairment was issued that established compensatory measures for exceeding the transient combustibles limits. The inspectors questioned the Shift Manager on the requirements to identify and control transient combustibles. The Shift Manager, after consulting with the fire protection coordinator, issued a fire impairment for the transient combustibles in room 59.

Analysis. The inspectors evaluated the safety significance of the finding. This finding affected the mitigating systems cornerstone and was considered more than minor since it affected the cornerstone attribute of Protection Against External Factors. Based on Manual Chapter 0609, Appendix F, the finding was determined to be in the Fire Prevention and Administrative Controls category because it affected the administrative controls used in fire prevention. The inspectors concluded that the degradation rating of the finding was low. This was based on the materials being stored in a room with no heat source and the materials not containing combustible liquids or being self-heating. The finding was characterized under the significance determination process as having very low safety significance (Green) since the degradation rating was low. Based on previous opportunities for personnel to recognize this condition, a human performance aspect was identified for this finding.

Enforcement. Technical Specification 5.8.1 requires, in part, that written procedures shall be established and maintained for implementation of the fire protection program. Procedure SO-G-102, "Fire Protection Program Plan," Revision 6, was the governing document for all fire protection program plan implementing procedures and references Procedure SO-G-91, "Control and Transportation of Combustible Materials," Revision 20, which implements fire protection requirements. Procedure SO-G-91, requires, in part, that relief from the general requirements for control of combustible materials or transient combustible limits may be obtained by submittal of Form FC-1244 (fire impairment) to the Fire Protection Engineer. Contrary to the above, Form FC-1244 (fire impairment) was not submitted to the fire protection engineer for the temporary storage of the steam generator nozzle dams containing rubber diaphragms and hoses that exceeded the 100 pound limit for transient combustibles in Room 59. This violation of Technical Specification 5.8.1 is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2005002-02). This violation is in the licensee's corrective action program as Condition Report 200500660.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the Probabilistic Risk Assessment Summary Notebook for internal flooding events. The inspectors performed walkdowns of the areas containing the temporary spent fuel pool cooling system to verify that safety-related equipment was not subject to damage as a result of internal flooding from the temporary system. The inspectors reviewed the internal flooding analysis that demonstrated that safety-related equipment in other rooms was not vulnerable to this internal flooding.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On January 31, 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 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)

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):

- Auxiliary Building Ventilating and Air Conditioning System Condensing Units, VA-95 and VA-96

- Intake Structure Sump Pumps, VD-2A and VD-2B

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:

- Outage of Turbine-Driven Auxiliary Feedwater Pump FW-10 and Main Feedwater Pump FW-4A on January 28, 2005

- Outage of Containment Spray Pump SI-3A, Low Pressure Safety Injection Pump SI-1A, and Emergency Diesel Generator 1 on February 16, 2005
- Outage of permanently installed spent fuel pool cooling system (replaced by the temporary spent fuel pool cooling system) on March 15, 2005
- Outage of 345 Kv electrical supply lines (replaced by the 161 Kv electrical supply lines) on March 24, 2005

b. Findings

No findings of significance were identified.

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

Plant Shutdown and Manual Reactor Trip

a. Inspection Scope

On February 25, 2005, operators initiated a plant power reduction at approximately 2 percent power per hour in preparation for a refueling outage that was scheduled to begin at 9 p.m. on February 26. The operators stopped the reduction at 12 percent power level. The operators opened the turbine generator output breakers, tripped the turbine generator, and prepared for turbine generator testing prior to reactor shutdown. At this time, a turbine generator operator noticed that Intermediate Stop Valve ISV-2 did not open as expected. Maintenance personnel were notified and asked to investigate the failure.

Meanwhile, the operators had indications of high/low steam generator level alarms and had difficulty maintaining the prescribed temperature band of +/- 0.2EF. The control room supervisor instructed the operator to place Bypass Feedwater Control Valves HCV-1105 and HCV-1106, which were in automatic, in manual mode to aid in controlling steam generator levels. The operator controlled the reactor coolant system temperature using the steam dump bypass valves.

Approximately 2 minutes later, the feedwater and temperature control operator reduced the Steam Generator B bypass control valve to 15 percent open from 30 percent, since level began to rise. After 1 minute the operator began to fill Steam Generator A. Another minute lapsed when the primary operator noticed pressurizer pressure and level lowering and informed the control room supervisor. The crew identified that reactor coolant system temperature was lowering and reactor power was rising due to feeding the steam generators. The reactor protection system loss of load trip automatically enabled and the reactor automatically tripped.

The crew entered Procedure EOP-00, “Standard Post Trip Actions,” Revision 17, following the reactor trip. With indications of an uncontrolled cooldown, the operators emergency borated and closed Main Feedwater Isolation Valves HCV-1385 and HCV-1386. This action terminated the feedwater induced cooldown and the operators transitioned to Procedure EOP-01, “Reactor Trip Recovery,” Revision 8, with all safety functions verified. When the operators completed Procedure EOP-01, they entered Procedure OP-3A, “Plant Shutdown,” Revision 55. The inspectors observed aspects of the plant downpower and the reactor trip.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability evaluations (four 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:

- Turbine-Driven Auxiliary Feedwater Pump FW-10, potential long-term inoperability due to condensate accumulation during loss of condenser vacuum (Condition Report 200500484)

- Lower portion of the reactor vessel head, paint/coatings peeling (Condition Report 200501523)

- Safety Injection Tanks SI-6A-D, Fill/Drain Line Relief Valve to RCDT WD-1, and SI-222, leaking by seat (Condition Report 200500434)

- Secondary air start pressure low light on Diesel Generator 1 did not light on low air pressure conditions (Condition Report 200500608)

b. Findings

.1 Long-Term Inoperability of Turbine-Driven Auxiliary Feedwater Pump

Introduction. A Green NCV of 10 CFR Part 50, Appendix B, Criterion III, was identified. The regulation requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to maintain design control of the turbine-driven auxiliary feedwater pump to ensure turbine casing condensate drains would function during accident conditions involving loss of condenser vacuum.

<u>Description</u>. The turbine-driven auxiliary feedwater pump (FW-10) is a Coffin turbo pump that provides cooling to the core via the steam generators during accident conditions. The pump is located in the auxiliary building at the 989' elevation. Steam admitted to the turbine is exhausted to the outside environment. To maintain the steam supply line warm while in a standby condition, a controlled amount of bypass steam is allowed to flow past the normally closed steam admission valves (YCV-1045A and -B) up to the normally closed Valve YCV-1045. Steam traps located at three different locations in the steam line downstream of this valve ensure that condensate is removed and will not impact the operability/functionality of the pump. These steam traps exhaust to the main turbine condenser at the 1007' elevation.

On February 7 and 8, 2005, the inspectors questioned whether FW-10 could perform its design function during a loss of condenser vacuum, given that the pump was located at a lower elevation than the connections to the condenser. The inspectors discussed with plant management and the operations crew that the pump may not be functional given that, without condenser vacuum to create a siphon, the accumulated condensate water would potentially flood the casing of the pump. The inspectors noted that, during a postulated loss of offsite power event, the circulating water pumps wouldn't be available, resulting in a loss of condenser vacuum. Following these discussions, on February 8, 2005, the licensee declared the component inoperable, entered Technical Specification 2.5, and made Event Notification Report 41386 to the NRC in accordance with 10 CFR 50.72 (v)(D).

The inspectors also noted that there had been seat leakage past Steam Admission Valve YCV-1045 since 2001, which was documented in Condition Report 200103716. Several condition reports had been written on the leaking Valve YCV-1045 since that time. The inspectors considered this to be a missed opportunity to identify a condition adverse to quality, specifically, the potential affect on the Coffin turbine due to condensate accumulation.

The licensee took the immediate corrective action to route the steam from Auxiliary Feedwater Pump Steam Trap ST-16 to the adjacent floor drain. This equipment alignment ensured that collecting condensate would be discharged and not accumulate within the pump casing while in a standby condition.

Fort Calhoun Updated Safety Analysis Report (USAR) Section 9.4.6 stated, "In the event of a loss of all ac power, the turbine-driven auxiliary feedwater pump would still be operational and would supply water to the steam generators from the emergency feedwater storage tank." Further, USAR Section 14.6, "Loss of Coolant Flow Incident," recognized a loss of off-site power as a design basis accident. The licensee subsequently performed an analysis of the operability and functionality of the pump as documented in Investigation Report FCE-2005-1, dated March 21, 2005. The analysis demonstrated the operability of the component by: (1) listing instances where Valve FW-10 had historically been started successfully without condenser vacuum, (2) engineering calculations showing acceptable stress levels on the turbine blades given partial submergence of the rotating element, and (3) detailed visual examination of the turbine rotor and blades with no indication of damage or degradation.

Analysis. The inspectors evaluated the safety significance of the finding. The performance deficiency was a failure to translate the design basis of the plant to maintain the function of the auxiliary feedwater system during a loss of offiste power or other event causing a loss of condenser vacuum. This finding was more than minor because it affected the Design Control Attribute of the Mitigating System Cornerstone as described in Appendix B of Inspection Manual Chapter 0612. Based on previous opportunities to recognize and correct this condition, a problem identification and resolution aspect was identified for this finding. The issue screened out as a Green finding because it was a design or qualification deficiency that was confirmed not to result in a loss of function as defined by NRC Generic Letter 91-18.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the measures established to assure that design information was correctly translated into procedures were inadequate. Specifically, the design basis function of the turbine-driven auxiliary feedwater pump during a loss of offsite power was not translated into specifications, drawings, procedures, and instructions. This violation of 10 CFR Part 50, Appendix B, Criterion III, is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2005002-03). This violation was entered into the licensee’s corrective action program as Condition Report 200500484.

.2 Degraded Coatings on Reactor Vessel

Introduction. A Green NCV of 10 CFR Part 50, Appendix B, Criterion V, was identified. The regulation requires that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee’s containment protective coatings procedure did not contain appropriate criteria to inspect the condition of safety-related coatings.

Description. On March 21, 2005, the inspectors performed an inspection of the underside of the reactor vessel. An inspector accompanied licensee personnel that were performing a walkdown to look for boric acid as required by Section XI of ASME Boiler and Pressure Code as required by Technical Specification Section 3.3.1. While no boric acid was observed, the inspector noted that large amounts of a coating on the lower reactor vessel had cracked and delaminated from the base metal, some of which had fallen away. The condition affected the entire lower portion of the reactor vessel to varying degrees.

The inspectors determined that the coating was applied by the manufacturer, Combustion Engineering, at the time of manufacture of the reactor vessel. The coating had been applied to other components (e.g., steam generators, pressurizer, etc.). The licensee subsequently performed an analysis that determined the following:

- Paint chips of concern were denser than postaccident water and would settle rather than float
- An insignificant amount of material was available for transport (approximately 0.5 ft^3, conservatively assuming 4 mil paint thickness)
- Previously performed asbestos abatement had already removed most of the paint on other major components (e.g., the steam generators and pressurizer)

The inspectors noted that the licensee's program document "Nuclear Safety Related Coatings" required that the scope of the program should include "Exposed carbon steel surfaces of equipment and components inside Containment . . ." Further, the program document required that "The inspection will be performed on a representative portion of each item to the extent that a conclusion can be reached as to the overall condition of the coatings."

The inspectors also noted that there had been a previous opportunity to identify the degraded coating. During the fall 2003 refueling outage, the licensee had performed a visual examination of the lower portion of the reactor vessel to look for leakage. The licensee had visually observed the condition of the lower reactor vessel and had noted in Condition Report 200304502 "paint flaking all around the surface of the bottom of the reactor vessel, the flakes range in size from 2 to 3 inches . . ." The inspectors considered this to be a missed opportunity to identify a condition adverse to quality.

<u>Analysis</u>. The inspectors evaluated the safety significance of the finding. This finding affected the Mitigating Systems cornerstone and was considered more than minor because it affected the procedure quality attribute of the cornerstone. Specifically, appropriate quantitative acceptance criteria was not provided to ensure that representative areas were selected for review within the coatings program. The finding was characterized under the significance determination process as having very low safety significance because the as-found reactor vessel head paint condition did not challenge the debris loading assumptions of the containment sumps and no actual loss of safety function occurred. Based on previous opportunities to recognize and correct this condition, a problem identification and resolution aspect was identified for this finding.

<u>Enforcement</u>. Title 10 CFR Part 50, Appendix B, Criterion V, states, in part, that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, the licensee failed to assure that procedures include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to assure that Procedure SE-PM-AE-1000, "Containment Protective Coatings Inspection," Revision 1,

contained appropriate acceptance criteria for which components should be inspected. This violation of 10 CFR Part 50, Appendix B, Criterion V, is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2005002-04). This violation is in the licensee's corrective action program as Condition Report 200501523.

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 cumulative effects of the workarounds (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 inspectors reviewed the deficiencies 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.

1R19 Postmaintenance Tests (71111.19)

a. Inspection Scope

The inspectors observed and/or reviewed postmaintenance tests (four 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 195573-02, troubleshoot manual rod insertion for Rod RC-10-23 on January 14, 2005

- Work Order 136566-02, temporary spent fuel pool cooling system installation/removal on February 28, 2005

- Work Order 175994-01, clean, inspect, lubricate, and adjust Manually-Operated Tap Disconnect Switch DS-T1A-1 on March 28, 2005

- Work Order 175631-01, replace Filter Regulator IA-HCV-2603B, Safety Injection Tanks SI-6A-D supply inboard isolation valve on April 5, 2005

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the licensee's refueling outage shutdown risk assessment to verify that the licensee appropriately considered risk in planning and scheduling the outage activities. The inspectors observed the reactor plant shutdown, the lowering of reactor coolant system water level to midloop conditions, core fuel off-load, shutdown maintenance activities, fuel inspections, and incore instrument insertions. The inspectors also performed several containment tours. The inspectors verified that the 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. At the end of the inspection period, the plant remained in a shutdown condition with all fuel off-loaded to the spent fuel pool.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- Procedure IC-ST-IA-3009, "Operability Test of IA-YCV-1045-C and Close Stroke Test of YCV-1045," on January 27, 2005

- Procedure SE-ST-MS-3005, "Main Steam Safety Valves Set Pressure Testing Using Furmanites's Trevitest Equipment," on February 22, 2005

- Procedure SE-ST-SI-3016, "Safety Injection Tank Discharge Check Valves Exercise Test," on March 11, 2005

- Procedure SE-ST-SI-3005, “Measurement of Post RAS Leakage to the SIRWT,” on March 12, 2005

- Procedure GM-PM-MX-03000, “Defuel, Clean, Inspect, and Refuel Diesel Fuel Oil Storage Tanks (FO-1 and FO-10),” on March 23, 2005

- Procedure IC-ST-AE-3139, “Type C Local Leakage Rate Test of Penetrations M-39 and M-53,” on March 31, 2005

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Modification EC 35462 (one inspection sample) that installed a temporary spent fuel pool cooling system to replace the permanently installed system. This system provided cooling capability to the spent fuel pool while the component cooling water and raw water systems were out of service for maintenance. The inspectors reviewed the associated 10 CFR 50.59 evaluation to confirm that the modification had no adverse impact on safety by introducing unanalyzed failure modes (e.g., adverse weather affecting the externally housed chiller units).

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Observation (71114.06)

a. Inspection Scope

On January 18, 2005, the inspectors observed an emergency preparedness drill from the simulator and the technical support center (one inspection sample). The purpose of the observation was to evaluate operator performance, licensee event classification, notification of state and local authorities, and the adequacy of protective action recommendations. The inspectors attended the licensee’s postdrill critiques and discussed observations with licensee management.

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 inspector used the requirements in 10 CFR Part 20, the Technical Specifications, and the licensee's procedures required by Technical Specifications as criteria for determining compliance. During the inspection, the inspector interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspector 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 the Containment Building, Refueling Floor, Auxiliary Building, Turbine Building, and Radwaste Building radiation, high radiation areas, and airborne radioactivity areas

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

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

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

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

- Self-assessments and audits 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 controls during job performance

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

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

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

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

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

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Licensee event reports (LERs) and special reports related to the access control program since the last inspection

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem CEDE (committed effective dose equivalent)

The inspector completed 21 of the required 21 samples.

b. Findings

.1 Introduction. The inspector identified a Green, self-revealing, NCV of Technical Specifications 5.11.1 and 5.11.2 involving the failure to control a high radiation area with dose rates in excess of 1,000 millirem per hour (restricted high radiation area).

Description. On March 4, 2005, a worker unexpectedly received an electronic dosimeter dose rate alarm during an entry into an unposted restricted high radiation area. Subsequently, the radiation protection staff found that dose rates around Valve RC-163 (RCS Loop-1 Hot Leg Drain Valve) near the Steam Generator A bay were 1,500 to 2,000 millirem per hour at 30 centimeters. The radiation protection technicians immediately posted, locked, and controlled the area as a restricted high radiation area.

The licensee's investigation revealed that changing operational conditions likely affected the location of a radiation source, and that the radiation protection department did not inform its technicians of this potential. In addition, based on historical radiation survey data from October 1999 to September 2003, the radiation protection department knew that the area around Valve RC-163 routinely had dose rates that were in excess of 1,000 millirem per hour at 30 centimeters. However, the radiation protection department had not communicated the significance of the historical data to its staff.

Analysis. The failure to control a restricted high radiation area in accordance with Technical Specifications 5.11.1 and 5.11.2 is a performance deficiency. This finding is greater than minor because it is associated with the occupational radiation safety exposure control attribute and affected the cornerstone objective, which is to ensure adequate protection of the worker health and safety from exposure to radiation. This occurrence involved a worker's unplanned, unintended dose or potential for such a dose that could have been significantly greater as a result of a single, minor, reasonable alteration of circumstances. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined that the finding was of very low safety significance because it did not involve: (1) ALARA (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 also had crosscutting aspects associated with human performance. The radiation protection organization did not inform its technicians about changing radiological conditions in the area of Valve RC-163 due to plant operations and based on historical data.

Enforcement. Technical Specifications 5.11.1 and 5.11.2 requires, in part, that areas accessible to personnel with dose rates in excess of 1000 millirem per hour at 30 centimeters from the source of radiation be provided with locked doors to prevent unauthorized entry. For individual high radiations areas accessible to personnel that are located within large areas, such as containment, or areas where no enclosure exists for purposes of locking and no enclosure can be reasonably constructed around the individual area, then that area shall be roped off, conspicuously posted, and have a flashing light as a warning device.

The failure to barricade, post, and control this restricted high radiation area in the Steam Generator A bay was a violation of Technical Specifications 5.11.1 and 5.11.2. This failure to control a restricted high radiation area is a finding of very low safety significance, and it has been entered into the licensee's corrective action program as Condition Report 200500977. This violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2005002-05).

.2 Introduction. The inspector identified an NCV of 10 CFR 20.1501(a) because the licensee's radiation protection staff failed to perform an adequate survey to evaluate radiological hazards when the particulate, iodine, and noble-gas radiation monitor (PING-211), located outside of the main containment hatch in Corridor 26 of the auxiliary building, alarmed.

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., but the radiation protection staff did not evaluate the radiological conditions in the area. The radiation protection staff had determined, in error, that the alarm was due to noise and reset the radiation monitor. The inspector observed that PING-211 had a chart with three channels that recorded and trended each type of radioactivity (particulate, iodine, and noble gas) on paper. The inspector further observed that an increasing particulate radioactivity trend appeared prominent on the chart paper the radiation monitor as early as 3:30 a.m., and there were two sets of noble gas spikes which coincided with the monitor alarms. However, the inspector determined that radiation protection technicians failed to evaluate the chart information. 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.

The inspector asked senior radiation protection technicians what radiation protection procedures were to be used in response to an alarming radiation monitor. The technicians were not able to identify a specific radiation protection procedure. They did identify procedure Standing Order SO-G-101, “Radiation Worker Practices,” Revision 26, Section 5.9.1, which required, in part, that “If any installed or portable radiological monitor alarms, immediately leave the area and notify the Control Room.” Radiation protection management later identified to the inspector that radiation protection Procedure RP-203, “Air Sample Collection and Analysis,” Section 7.1.1 stated, in part, that “Particulate, radio iodine, or noble gas samples, as appropriate, should be taken when activity is suspected.” The inspector concluded that the radiation protection department did not have any specific requirements for responding to alarming PING radiation monitors. On March 18, 2005, radiation protection management issued written instructions to radiation protection staff for responding to alarming continuous air monitors.

On March 18, 2005, the licensee identified the likely cause of the airborne radioactive material (Co-60) in the containment and auxiliary buildings. On March 17, 2005, at 4:06 a.m. the operations department had shut down the containment building ventilation system. Later that morning when the operations department restarted the Containment Building ventilation system at approximately 5 a.m., they started the supply fans before starting the exhaust fans. Therefore, contamination from the refueling floor and other areas of the containment building went airborne and contaminated the auxiliary building through the personnel hatch.

Analysis. The failure to perform an adequate radiation survey is a performance deficiency. The finding is more than minor because it affected the Occupational Radiation Safety cornerstone objective to protect worker health and safety from radiation and radioactive materials. This finding was associated with the cornerstone attribute of exposure control 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) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding also had crosscutting aspects associated with human performance. The radiation protection organization did not have an effective process for its technicians to evaluate potential radiological hazards associated with alarming airborne radiation monitors.

Enforcement. Pursuant to 10 CFR 20.1003, "survey" means an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. Title 10 CFR 20.1501(a) requires, in part, that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with, in regard to regulations in 10 CFR Part 20, and that are reasonable under the circumstances to evaluate the extent of radiation levels and the potential radiological hazards that could be present. Title 10 CFR 20.1201 requires, in part, that the licensee control the occupational dose to individual adults to the dose limits in 10 CFR Part 20. The inspector determined that the licensee's failure to survey contributed to 11 workers receiving unplanned and unintended occupational exposure (less than 5 millirem) from airborne Co-60.

Because the failure to perform a radiation survey resulted in an occurrence of very low safety significance, and it has been entered into the licensee's corrective action program as Condition Report FCS-2005-01401, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2005002-06).

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspector sampled licensee submittals for the performance indicators listed below from October 2004 to February 2005. To verify the accuracy of the performance indicator data reported during that period, performance indicator definitions and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

Occupational Radiation Safety Cornerstone

Occupational Exposure Control Effectiveness Performance Indicator

Licensee records reviewed included corrective action documentation that identified occurrences of restricted 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 99-02). 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, restricted high radiation, and very high radiation areas were properly controlled.

Public Radiation Safety Cornerstone

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

The inspector reviewed radiological effluent release program corrective action records and annual effluent release reports documented from October 2004 to February 2005 to determine if any liquid or gaseous effluent releases resulted in events that exceeded the performance indicator thresholds. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Selected Issue Followup Inspection

a. Inspection Scope

The inspectors selected seven condition reports for detailed review (200302602, 200302607, 200302623, 200400244, 200400517, 200400840, and 200401755). The condition reports were associated with problems associated with the diesel generators. The reports were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized.

b. Findings and Observations

No findings of significance were identified.

.2 Identification and Resolution Issues Documented in the Current Inspection Report

Please refer to Sections 1R15.1 and 1R15.2 of this report for descriptions of missed opportunities to identify conditions adverse to quality, associated with Pump FW-10 design and reactor vessel head paint peeling. The inspector reviewed corrective action documents for root cause/apparent cause analysis against the licensee's problem identification and resolution process. No findings of significance were identified.

Section 2OS1 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding access controls to radiologically significant areas and radiation worker practices. The inspector reviewed corrective action documents for root cause/apparent cause analysis against the licensee's problem identification and resolution process. No findings of significance were identified.

4OA3 Event Followup (71153)

.1 (Closed) LER 05000285/2003003-00, Reactor Trip During Plant Shutdown due to Inadequate Preparation

On September 12, 2003, during a planned reactor plant shutdown, the reactor was tripped manually because Axial Shape Index (ASI) could not be maintained within the band specified by plant management. Because the ASI guidance was not documented, this event constituted an unplanned automatic or manual reactor trip, hence, was reportable in accordance with 10 CFR 50.73 (a)(2)(iv)(A). The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the event in Condition Report 200303492. This LER is closed.

.2 (Closed) Notice of Violation 05000285/2004002-02, Inadequate Diesel Generator Surveillance Test Procedure Acceptance Criteria

On May 14, 2004 a Notice of Violation was issued along with NRC Inspection Report 05000285/2004002 documenting a violation of 10 CFR Part 50, Appendix B, Criterion V. This was the result of the diesel generator test procedure not containing appropriate quantitative or qualitative acceptance criteria to determine operability of diesel generators when conducting the full speed starts of the diesel generators. The licensee's acceptance criteria did not account for a 2 hertz speed droop of the fully loaded diesel generator when selecting the minimum acceptable frequency. The licensee had previously received an NCV (NCV 05000285/2003005-02) as a result of a similar condition adverse to quality.

The inspectors reviewed the licensee's response to the Notice of Violation, condition reports documenting the occurrence, and diesel generator surveillance test procedures. Licensed operators and engineering personnel were interviewed on their understanding of the surveillance procedures. The inspectors observed the conduct of Surveillance Procedure OP-ST-DG-0001, "Diesel Generator 1 Check," Revision 46, to verify that the correct data was being obtained and evaluated against the appropriate acceptance criteria.

.3 (Closed) Notice of Violation 05000285/200309-01, Failure to follow radiation protection procedural and radiation work permit requirements

The inspector reviewed the licensee's response to Notice of Violation 5000285/200309-01, NCV 05000285/2004005-02, and associated condition reports documenting subsequent similar occurrences. The inspector interviewed cognizant radiation protection and security personnel. The inspector reviewed the licensee's corrective actions to determine if they were adequate to prevent recurrence of the violations.

NRC Inspection Report 05000285/2003009 and the Notice of Violation documented that 30 security officers on 62 occasions, between April 27 and October 8, 2002, did not obtain electronic alarming dosimeters and did not sign in on the required radiation work permit before entering a radiologically controlled area, in violation of Technical Specification 5.8.1. During an inspection on October 8, 2004, an inspector found four additional examples of a violation of Technical Specification 5.8.1(a) had occurred because security personnel failed to follow radiation protection procedural and radiation work permit requirements.

The inspector reviewed and observed the licensee's corrective actions that were implemented per Condition Reports 200402903, 200403073, and 200403507 and determined that the actions appeared adequate to prevent recurrence of the violations. The principal corrective action was the licensee's installation of a turnstile at the entrance of the radiologically controlled area.

4OA4 Crosscutting Aspects of Findings

Section 2OS1 described violations involving a failure to control a restricted high radiation area and failure to perform an adequate radiation survey had crosscutting aspects associated with human performance in radiation safety.

4OA5 Other Activities

Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (TI 2515/160)

a. Inspection Scope

On May 28, 2004, the NRC issued Bulletin 2004-01, "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors." The purpose of this bulletin was to: (1) advise PWR licensees that current methods of inspecting Alloy 82/182/600 materials used in the fabrication of pressurizer penetrations and steam space piping connections may need to be supplemented with additional measures to detect and adequately characterize flaws due to primary water stress corrosion cracking (PWSCC); (2) request pressurized water reactor addressees to provide the NRC with the information related to the materials from which the pressurizer penetrations and steam space piping connections at their facilities were fabricated; and (3) request pressurized water reactor licensees to provide the NRC with the information related to the inspections that have been and those that will be performed to ensure that degradation of Alloy 82/182/600 materials used in the fabrication of pressurizer penetrations and steam space piping connections will be identified, adequately characterized, and repaired.

The objective of Temporary Instruction (TI) 2515/160, “Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors,” was to support the NRC review of licensees’ activities for inspecting pressurizer penetrations and steam space piping connections made from Alloy 82/182/600 materials and to determine whether the inspections of these components are implemented in accordance with the licensee responses to Bulletin 2004-01. In response to Bulletin 2004-01, the licensee stated that they perform a bare metal visual inspection of 100 percent of all pressurizer heater sleeve locations, during each outage. The inspectors performed a review, in accordance with TI 2515/160, of the licensee’s procedures, equipment, and personnel used for pressurizer penetration nozzles and steam space piping connections examinations to confirm that the licensee met commitments associated with Bulletin 2004-01. The results of the inspectors’ review included documenting observations and conclusions in response to the questions identified in TI 2515/160.

b. Observations

Summary: Based upon a bare metal visual examination of the pressurizer, the licensee did not identify any indications of boric acid leaks from pressure retaining components in the pressurizer system.

Evaluation of Inspection Requirements

In accordance with the requirements of TI 2515/160, inspectors evaluated and answered the following questions:

1. For each of the examination methods used during the outage, was the examination performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

   Yes. The licensee conducted a direct visual examination of the bare metal surface of the lower pressurizer head heater penetration nozzles and ultrasonic testing of pressurizer nozzles with knowledgeable staff members certified to Level II as VT-2 examiners in accordance with Procedure QCP-200, “Certification Requirements of Quality Control Inspectors,” Revision 24. This qualification and certification procedure referenced the industry standard ANSI/ANST CP-189, “Standard for Qualification and Certification of Nondestructive Testing Personnel.”

2. For each of the examination methods used during the outage, was the examination performed in accordance with demonstrated procedures?

Yes. The inspectors observed the licensee inspector performing the bare metal inspection of the vessel head in accordance with Procedure QCP-400, “Visual Inspection,” Revision 6. The licensee considered this procedure to be demonstrated because examination personnel could resolve lower case alpha numeric characters 0.158 inch in height at a distance of 6 feet.

3. For each of the examination methods used during the outage, was the examination able to identify, disposition, and resolve deficiencies, capable of identifying and characterizing boron deposits, and capable of identifying leakage in pressurizer penetration nozzle or steam space piping components, as discussed in NRC Bulletin 2004-01?

   Yes. The inspectors concluded that the licensee’s direct visual examinations were capable of detecting leakage from cracking in pressurizer penetrations if it had existed. This conclusion was based upon the inspectors' direct observations of pressurizer penetration locations which were free of debris or deposits that could mask evidence of leakage in the areas examined.

   Because the licensee did not identify any deposits indicative of leakage in the areas examined, the inspectors could not assess the licensee’s plans to characterize leakage on pressurizer components. However the inspectors, during a direct observation of the pressurizer in order to verify the licensee’s results, identified what appeared to be several boron accumulations. The licensee performed chemistry analysis of samples taken at these locations and confirmed that there was no boron present (insulation).

4. What was the physical condition of the penetration nozzle and steam space piping components in the pressurizer system and what, if any, impediments were identified (e.g., debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

   The Fort Calhoun Station pressurizer lower head was covered with asbestos blocks and blanket insulation. This material was removed to allow installation of staging directly below the lower pressurizer head. The staging platform provided sufficient access to perform the bare metal examination of the heater penetrations. The inspectors performed a direct visual inspection for portions of each of the 72 pressurizer heater penetration nozzles. Based on this examination, the lower pressurizer head and penetration nozzle area was clean and free of debris, deposits, or other obstructions which would mask evidence of leakage. As able, the inspectors performed visual inspections of upper pressurizer head penetrations and steam space piping connections.

5. How was the visual examination conducted and how complete was the coverage (e.g., video camera or direct visual by examination personnel)?

The licensee was able to perform a direct visual examination for 360 degrees around each weldment or penetration. This examination included each of 72 pressurizer heater penetrations, and welds for the safety valves, level piping penetration, power-operated relief valve line, and temperature element.

6. What material deficiencies were identified that required repair?

   None.

7. If volumetric or surface examinations were used for the augmented inspection examinations, what process did the licensee use to evaluate and dispose of any indications that may have been detected as a result of these examinations?

   Not applicable. No augmented volumetric or surface examinations were performed.

8. Did the licensee perform appropriate followup examinations for indications of boric acid leaks from pressure retaining components in the pressurizer system?

   Not applicable. The licensee did not identify any indications of boric acid leaks from pressure retaining components in the pressurizer system.

c. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

.1 On March 18, 2005, the inspector presented the inspection results to Mr. M. Frans, Assistant 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.

.2 On April 11, 2005, the inspectors presented the inspection results to Mr. D. Bannister, Plant Manager, and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information that was used during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bannister, Plant Manager
A. Clark, Manager, Security and Emergency Planning
H. Faulhaber, Manager, Work Management
M. Frans, Assistant Plant Manager
S. Gebers, Corporate Health Physicist
W. Goodell, Manager, Operations
R. Haug, Manager, Chemistry
J. Herman, Manager, Nuclear Licensing
T. Maine, Supervisor, Radiological Operations
E. Matzke, Station Licensing Engineer
R. Phelps, Division Manager, Nuclear Engineering
T. Pilmaier, Manager, Corrective Action Group
M. Puckett, Manager, Radiation Protection
M. Tesar, Division Manager, Nuclear Support
J. W. Tillis, Manager, Maintenance
R. Westcott, Manager, Training

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000285/2005002-01	NCV	Failure to Ensure that Fire Barriers Protecting Safety-Related Areas Were Functional (Section 1R05b1)
05000285/2005002-02	NCV	Failure to Control Transient Combustible Materials that Exceeded the Fire Load limit for an Area (Section 1RO5b2)
05000285/2005002-03	NCV	Failure to Translate Design Basis of the Turbine Driven Auxiliary Feedwater Pump into Procedures (Section 1R15.1)
05000285/2005002-04	NCV	Failure to Include Quantitative Acceptance Criteria for Containment Protective Coatings Inspection (Section 1R15.2)
05000285/2005002-05	NCV	Failure to control a restricted high radiation area per Technical Specifications 5.11.1 and 5.11.2 (Section 2OS1)
05000285/2005002-06	NCV	Failure to perform an adequate survey to evaluate radiological hazards per 10 CFR 20.1501 (Section 2OS1)

Closed

05000285/2004002-02	NOV	Inadequate Diesel Generator Surveillance Test Procedure Acceptance Criteria (Section 4OA5)
05000285/2003009-01	NOV	Failure to follow radiation protection procedural and radiation work permit requirements (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Procedures:

EOP-2, "Loss of Off-Site Power/Loss of Forced Circulation," Revision 12
EOP-7, "Station Blackout," Revision 8
AOP-36, "Loss of Spent Fuel Pool Cooling," Revision 3

Analyses:

EA FC-93-23, "Containment Interior Coatings Integrity," Revision 0

EA FC-05-09, "Effect of Paint Chips from the Reactor on Containment Sumps," Revision 0

FC 06011, "Seismic Qualification of Fuel Oil Storage Tanks FO-1 and FO-10," Revision 0
FC 06932, "Spent Fuel Pool Heat Load Following Cycle 22 Full Core Offload," Revision 0

FC 07064, "Heat Removal Analysis for Spent Fuel Pool Temporary Cooling System," Revision C

EC 35462, "OI-SFP-7/Alternate Spent Fuel Pool Cooling Using Chillers," dated February 11, 2005

EC 35918, "PE-OT-SFP-0001/Temporary Alternate Spent Fuel Pool Cooling System Installation/Removal," dated February 14, 2005

EC 35893, "Change Normal Operating Positions for Valves MS-242, MS-244, and MS-246," dated February 8, 2005

"Investigation of Coffin Turbo Pump 8494-DEB-50-177-183 Auxiliary Feedwater Service FW-10," by Foster Cove Engineering, Inc. dated March 21, 2005

Condition Reports:

200103716
200304502
200500357
200500590
200501068
200501196
200501222
200501564
200501607

Miscellaneous:

Technical Specifications 3.5, “Containment Test”

Technical Specifications 4.4, “Fuel Storage”

USAR Section 2.4, “Site and Environs - Seismology”

USAR Section 5.2, “Protective Coatings and Paints Inside Containment”

USAR Section 9.4, “Auxiliary Systems”

Original Final Safety Analysis Report Section 9.6, “Spent Fuel Pool Cooling System,” no revision or date given

USAR Section 9.6, “Spent Fuel Pool Cooling System,” Revision 7

Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02

Drawing Number 3742, “Reactor Vessel Insulation Layout R2," Revision 1

Drawing Number 11405-N–252, “Flow Diagram Steam PI&D,” Revision 95

Figure F-2, “Response Spectra Maximum Hypothetical Earthquake”

Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” dated November 2000

Regulatory Guide 1.54, “Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants,” dated June 1973

50.59 Evaluation “OI-SFP-7, Temporary Spent Fuel Pool Cooling Via Chillers,” dated March 12, 2005

NEI 96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1

Work Document WO200007-01, “Remove FW-10's Turbine Housing Cover,” March 6, 2005

Work Document WO00176432, “Defuel, Clean, Inspect and Refuel Tank,” March 23, 2005

Fire Protection Impairments 2005075, 2005094, 2005077, 2005061

Control Room Logs from the following dates:
November 23-25, 1996
May 26 to June 4, 1998
November 8-10, 1999
April 26-28, 2001
October 23-31, 2003