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

November 14, 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 05000285/2005004 AND 07200054/2005001

Dear Mr. Ridenoure:

On September 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 October 11, 2005, with Mr. Harry Faulhaber, Senior Division Manager, Nuclear Engineering, and other members of your staff.

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

This report documents a finding that has a potential safety significance greater than very low safety significance. The issue concerns a finding on a nonfunctional backup air supply on the condenser makeup valve. This finding did represent an immediate safety concern. However, compensatory measures were established until the backup air supply was restored. In addition, the report documents two NRC-identified findings that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC determined that these findings are not associated with NRC requirements. However, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. NRC is treating this violation as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy because of the very low safety significance of the violation and because it is entered into your corrective action program. If you contest this noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document

Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator Region IV, the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Fort Calhoun Station.

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

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 and 72-054
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2005004 and 07200054/2005001
w/attachment: Supplemental Information

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SISP Review Completed: _DNG ADAMS: / Yes ☐ No Initials: _DNG_____
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RIV:RI:DRP/E	SRI:DRP/E	C:DRS/PSB	C:DRS/OB	C:DRS/EB1
LMWilloughby	JDHanna	MPShannon	ATGody	CJPaulk
T-DNGraves	***E-DNGraves***	***E-DNGraves***	***/RA/***	***/RA/***
11/08/05	11/09/05	11/09/05	11/9/05	11/9/05

C:DRS/EB2	C:DNMS/FCD	C:DRP/E	
LJSmith	DBSpitzberg	DNGraves	
E-DNGraves	***E-DNGraves***	***/RA/***	
11/09/05	11/9/05	11/14/05	

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285 and 72-054
License:	DPR-40
Report:	05000285/2005004 and 07200054/2005001
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	Fort Calhoun Station FC-2-4 Adm. P.O. Box 399, Highway 75 - North of Fort Calhoun Fort Calhoun, Nebraska
Dates:	July 1 through September 30, 2005
Inspectors:	J. Hanna, Senior Resident Inspector L. Willoughby, Resident Inspector Z. Dunham, Senior Resident Inspector, Columbia Generating Station P. Elkmann, Emergency Preparedness Inspector S. Atwater, Health Physicist L. Ricketson, Sr. Health Physicist T. Ahn, High Level Waste Reviewer, Office of High Level Waste and Repository Safety
Approved By:	David N. Graves, Chief, Project Branch E Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2005004 and 07200054/2005001; 07/01/2005 - 09/30/2005; Fort Calhoun Station, Integrated Resident and Regional Report; Operability Evaluations, ALARA Planning and Controls, and Crosscutting Areas.

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

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- TBD. The inspectors reviewed a self-revealing finding having potential safety significance greater than very low significance. The backup source of instrument air to the main condenser makeup Valve LCV-1190 was unavailable for approximately 19 days of power operation. The backup air supply ensures that the non-safety-related diesel-driven auxiliary feedwater pump, FW-54, has a source of water from the condensate storage tank. This finding had crosscutting aspects of problem identification and resolution in that although the condition was identified on multiple occasions, the significance was not evaluated or recognized for 19 days. This finding was entered into the licensee's corrective action program as Condition Report 200503231.

  The finding is unresolved pending completion of the significance determination process analysis. The finding is greater than minor because the condition impacted the reliability and availability of FW-54 to mitigate the consequences of a loss of instrument air or loss of offsite power. The finding was determined to have potential safety significance greater than very low significance because the finding represented an actual loss of safety function (during a loss of instrument air or loss of offsite power) of one or more of the non-technical specification trains or equipment designated as risk-significant (Section 1R15.b).

Cornerstone: Occupational Radiation Safety

- Green. The inspector identified a finding because performance deficiencies resulted in a collective dose for a work activity that exceeded five person-rem and 150 percent of the legitimate dose estimate. Radiation Work Permit 05-3530, "Reactor Vessel Head Inspection in Restricted High Radiation Areas," was estimated to require approximately three person-rem to complete, but actually accrued approximately 13.6 person-rem. The licensee used an unproven technology to inspect for defects. As a result, equipment problems caused the planned work duration and dose to be greatly exceeded. The project was poorly planned, poorly implemented, and poorly overseen by management. The finding was placed into the licensee's corrective action program as Condition Report 200501853.

This finding was more than minor because it was associated with the occupational radiation safety cornerstone attribute (al low as reasonably achievable planning/estimated dose) and affected the associated cornerstone objective in that it increased the collective dose. When processed through the occupational radiation safety significance determination process, this al low as reasonably achievable finding was found to have no more than very low safety significance because: (1) the finding was related to al low as reasonably achievable planning or work control, but (2) the licensee's 3-year rolling average collective dose was not greater than 135 person-rem. In addition, this finding had crosscutting aspects associated with human performance, in that the poorly managed project resulted in the finding (Section 2OS2).

- Green. The inspector identified a finding because the licensee did not adequately plan for emergent work, causing the collective dose for the work activity to exceed 5 person-rem and 150 percent of the legitimate dose estimate. Radiation Work Permit 05-3519, "SI-220 Valve Replacement in Restricted High Radiation Areas," was estimated to require approximately 3 person-rem to complete, but actually accrued approximately 9.8 person-rem. The licensee failed to formulate reasonably accurate dose estimates and plan dose saving measure for the emergent work after problems occurred. The finding was placed into the licensee's corrective action program as Condition Reports 200504080 and 200504274.

  This finding was more than minor because it was associated with the occupational radiation safety cornerstone attribute (al low as reasonably achievable planning/estimated dose) and affected the associated cornerstone objective in that it increased the collective dose. When processed through the occupational radiation safety significance determination process, this al low as reasonably achievable finding was found to have no more than very low safety significance because: (1) the finding was related to al low as reasonably achievable planning or work control, but (2) the licensee's 3-year rolling average collective dose was not greater than 135 person-rem. In addition, this finding had crosscutting aspects associated with human performance and problem identification and resolution. The ALARA planner's failure to take proper actions directly caused the finding. The work group failed to address problems that caused the unplanned dose through the corrective action program (Section 2OS2).

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 40A7 of this report.

REPORT DETAILS

Summary of Plant Status

The unit began this inspection period in Mode 1 at 96 percent reactor power. On July 5 reactor power was increased to 100 percent where the plant remained until the end of the inspection period.

1. REACTOR SAFETY

1R04 Equipment Alignments (71111.04)

1. Partial Equipment Walk-downs

a. Inspection Scope

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

- Diesel-driven and motor-driven auxiliary feedwater pumps while the turbine-driven auxiliary feedwater pump was secured for maintenance

- Diesel Generator 2 fuel oil system while Diesel Generator 1 was secured for maintenance

- Safety Related electrical Busses 1A3 and 1A4 while Diesel Generator 2 was secured for maintenance

b. Findings

No findings of significance were identified.

2. Complete System Walk-downs

a. Inspection Scope

The inspectors conducted a detailed review of the alignment and condition of the raw water system (one inspection sample). The inspectors reviewed open work orders and condition reports associated with the system. The inspectors performed a walk-down of accessible portions of the system. During the walk-down, inspectors verified that the system was properly aligned in accordance with piping and instrumentation Drawing 11405-M100, "Raw Water Flow Diagram," Revision 88 and Procedure OI-RW-1, "Raw Water System Normal Operation," Revision 68.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

1. Routine Fire Inspection Tours

a. Inspection Scope

The inspectors performed routine fire inspection tours (four 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 43 - Auxiliary Building Upper Level (Room 81)
- Fire Area 31 - Intake Structure (Upper and Lower Levels)
- Upper and Lower Warehouse Fire Areas
- Fire Area 46.2 - Turbine Building Elevations 990' through 1036'

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Re-qualification (71111.11)

a. Inspection Scope

The inspectors performed one licensed operator requalification observation (one inspection sample). On August 29, 2005, the inspectors observed licensed operator re-qualification training activities, including the licensed operators' performance and the evaluators' critique. The inspectors compared performance in the simulator with the Licensed Operator Training Template 84104d, "Loss of CCW to RCPs - Natural Circulation Required," Revision 1, and with performance observed in the control room during this inspection period. The focus of the inspection was on high-risk licensed operator actions, operator activities associated with the emergency plan, and previous lessons-learned items. These items were evaluated to ensure that operator performance was consistent with protection of the reactor core during postulated accidents.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

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

- Engineered Safety Features System Switch CS-A/LS
- Loop 2 to Shutdown Cooling Isolation Valve HCV-348

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 (five inspection samples) for equipment outages 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:

- Testing of safety injection loop injector Valves HCV-331 and HCV-333, venting the low pressure safety injection header, rebuilding charging Pump 1C, and reactor protection surveillance testing on July 18, 2005

- Elevated risk condition and associated compensatory measures while emergency diesel Generator 1 was removed from service for surveillance testing on August 3, 2005

- Elevated risk condition and associated compensatory measures while both the turbine and motor-driven auxiliary feedwater Pumps (FW-10 and FW-6) were removed from service for surveillance testing on August 4, 2005

- Elevated risk condition and associated compensatory measures while the turbine-driven auxiliary feedwater Pump (FW-10) was removed from service for routine preventive maintenance on August 10, 2005

- Elevated risk condition and associated compensatory measures while conducting emergency diesel Generator 2 surveillance testing, charging Pump 1C maintenance and emergent work on control room ventilation filter Unit VA-64B on August 17, 2005

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 licensee personnel. The inspectors reviewed the operability evaluations and cause assessments for the following:

- Condensate makeup control Valve LCV-1190 backup nitrogen supply inoperable for greater than 30 days (CR 200503231)

- Component cooling waters leak isolated to containment cooling and filter Unit VA-15A Cooling Coil VA-1A (CR 200503367)

- Low settings found for Pressurizer RC-4 relief isolation Valve HCV-151 instantaneous breaker trip setting (CR 200503877)

- Turbine-driven auxiliary feedwater Pump FW-10 back pressure trip Latch FW-64 inoperable during a calibration (CR 200504448)

b. Findings

Introduction. The inspectors reviewed a self-revealing finding associated with the unintended unavailability of the diesel-driven auxiliary feedwater Pump (FW-54) due to the backup source of Instrument Air (IA) to the main condenser makeup Valve LCV-1190 being nonfunctional. This issue is unresolved pending completion of the significance determination analysis for this issue.

Description. On May 14, 2005, while the plant was shutdown during a refueling outage, the licensee identified that the four nitrogen cylinders that provided a backup source of IA to the main condenser makeup Valve LCV-1190 were depressurized. These cylinders ensured that Valve LCV-1190, which fails open on a loss of instrument air (LOIA), would remain closed for up to four hours to prevent the condensate storage tank from draining to the condenser hotwell. This would provide operators time to manually isolate Valve LCV-1190 during a LOIA and ensure the diesel-driven auxiliary feedwater Pump FW-54 has a source of water. Pump FW-54 is a non-safety related component that has no associated technical specifications (TS) but is risk-important because it provides a redundant supply of auxiliary feedwater to the steam generators. In addition, Pump FW-54 can also provide makeup water to the emergency feedwater storage tank from the condensate storage tank. Because of the redundant capability of Pump FW-54, the licensee had credited the pump in their plant risk model and considered the pump to have a high risk significance function per 10 CFR 50.65. Work Request 82847 was written on May 14 noting that the cylinders were depressurized. On May 27, the cylinders were again noted to be empty and Work Request 83298 was written to address the issue. On June 5, 12, 19, and 26, equipment operators documented in the turbine building logs that the cylinders were reading zero psig. The minimum acceptable pressure noted on the log sheet was 1025 psig. Although following the June 5 log reading the equipment operator referenced that a work request had been initiated on May 14, no actions were taken to ensure that the cylinders were replaced until a shift manager recognized the potential significance of the depressurized cylinders on June 28 and initiated a condition report (CR 200503231).

The licensee performed a root cause analysis and determined that during the spring 2005 refueling outage the four nitrogen cylinders had become depressurized during a planned replacement of the main condenser tubes. Prior to June 28, 2005, equipment operators had noted (on June 5, 12, 19, and 26) in the turbine-building Log FC-78 that the nitrogen cylinder pressure was indicating zero psig with a minimum acceptable pressure of 1025 psig. However, following each of those instances of logging zero psig, the equipment operator failed to write a condition report to document the equipment deficiency. Additionally, the shift manager reviewed and failed to act on the deficient condition noted in the logs recorded on June 5, 12, 19, and 26, 2005. The failure to identify the issue during these reviews indicated that the log review process, which was designed to identify degraded equipment conditions and adverse trends, was not properly implemented in this case.

Standing Order SO-R-2, “Condition Reporting and Corrective Action,” Revision 30, Attachment 7.1 “Guidelines for Identification of Conditions to be Reported,” provided that equipment, material, or components available for operation in the plant that are determined to be, or may potentially be, nonconforming should be reported via the condition reporting system. The licensee’s failure to document in the corrective action system, or properly evaluate the significance of the as-found depressurized condition of the nitrogen cylinders is considered to be a performance deficiency that was reasonably within the licensee's ability to identify and prevent from occurring. The inspectors considered the finding to be greater than minor because the condition impacted the reliability and availability of Pump FW-54 to mitigate the consequences of a loss of instrument air and loss of offsite power events.

Analysis. Using Phase 1 of the significance determination process, the inspectors determined that the finding represented an actual loss of safety function during a loss of IA or loss of offsite power of one or more of the non-technical specification trains or equipment designated as risk-significant per 10 CFR 50.65 for greater than 24 hours. Consequently, a Phase 2 evaluation was required.

The inspectors used the Fort Calhoun site-specific worksheets and Phase 2 of the significance determination process to further evaluate the risk significance of the finding. The inspectors assumed that operator recovery credit was warranted because the operators were directed in Procedure AOP-17, "Loss of Instrument Air," Revision 7 to manually isolate FW-Valve LCV-1190 in the event that IA header pressure was less than 50 psig. The inspectors assumed an exposure time of 3-30 days during the evaluation because the functionality of Pump FW-54 was affected due to the backup air supply to Valve LCV-1190 being unavailable for approximately 19 days of power operations. In addition, the inspectors only evaluated the loss of instrument air and loss of offsite power worksheets since these were the only initiating events during which the functionality of Pump FW-54 was affected. At the end of the inspection period, the review of the Phase 2 process had not been completed.

The inspectors considered the finding to have crosscutting aspects related to problem identification and resolution. Although the depressurized nitrogen cylinders were identified and documented on multiple occasions, the condition was not properly documented in the corrective action program nor evaluated for significance until June 28, 2005.

Enforcement. The inspectors determined that Valve LCV-1190 and Pump FW-54 were not safety-related and were not required per TS. Therefore, no violation of regulatory requirements (e.g., TSs or failure to identify a Condition Adverse to Quality) was identified. The nitrogen bottles were subsequently replaced and verified to be at the required pressure so no safety issue currently exists. Pending determination of the final safety significance of this issue, this issue is being treated as an unresolved item (URI) 05000285/2005004-01, "Inoperable Backup Instrument Air to Condensate Makeup Control Valve."

1R19 Post-Maintenance Tests (71111.19)

a. Inspection Scope

The inspectors observed and/or reviewed post maintenance 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 00166019-01, replace SI-123 (SI-1B Drain) and cap leak-off line on August 24, 2005
- Work Order 00212119-01, refurbish or replace VA-46B Hot Gas Valve on September 2, 2005
- Work Order 00215400-02, replace grounded condenser fan motor for VA-46B on September 2, 2005
- Work Order 00217463-01, adjust AC-10D pump impeller lift to 0.050 inches on September 8, 2005

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 OP-ST-ESF-0010, “Channel B Safety Injection, Containment Spray and Recirculation Actuation Signal Test,” on July 7, 2005
- Procedure OP-ST-RC-3001, “Reactor Coolant System (RCS) Leak Rate Test,” on July 29, 2005
- Procedure IC-ST-RC-0003, “Monthly Functional Test of Pressurizer Level Instrument L-101X and L-101Y,” on July 29, 2005
- Procedure OP-ST-DG-0001, “Diesel Generator 1 Check,” on August 3, 2005
- Procedure IC-ST-IA-3003, “Raw Water Instrument Air Accumulator Check Valve Operability Test,” on August 8, 2005
- Procedure OP-ST-RW-3031, “AC-10D Raw Water Pump Quarterly In-service Test,” on September 7, 2005

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Modification EC 37080 for bypassing the test function on Turbine EHC Master Trip Solenoid Valves and its associated 10 CFR Part 50.59 screening (one inspection sample). The inspectors verified the modification had no effect on system operability or availability. The inspectors reviewed the post installation test results to confirm that the test was satisfactory and that there was no adverse impact of the temporary modification on the permanent system.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspectors performed in-office reviews of the following revisions to the Fort Calhoun Station Radiological Emergency Response Plan:

- Section D, “Emergency Classification System,” Revision 11
- Section F, “Emergency Communications,” Revision 15
- Section H, “Emergency Facilities and Equipment,” Revision 31
- Section L, “Medical and Public Health Support,” Revision 12
- Section M, “Recovery and Reentry Planning and Post-Accident Operations,” Revision 14
- Appendix D, “Emergency Response Plan Authority,” Revision 3

These revisions changed the plan and information format. These revisions were compared to their previous revisions, to the criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and

Preparedness in Support of Nuclear Power Plants," Revision 1, and to the requirements of 10 CFR Part 50.47(b) and Part 50.54(q) to determine if the licensee adequately implemented 10 CFR Part 50.54(q). The inspector completed one sample during the inspection.

b. Findings

No findings of significance were identified.

1EP6 Drill Observation (71114.06)

a. Inspection Scope

On July 19 and August 9, 2005, the inspectors observed aspects of the emergency preparedness drills from the simulator and the technical support center (two inspection samples). 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 post drill critiques and discussed the observation with licensee management.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

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

- Current 3-year rolling average collective exposure
- Eight work activities from previous work history data, which resulted in the highest personnel collective exposures.
- Site specific trends in collective exposures, plant historical data, and source-term measurements
- Site specific ALARA procedures

- Intended versus actual work activity doses and the reasons for any inconsistencies

- Person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements

- Post-job (work activity) reviews

- Assumptions and basis for the current annual collective exposure estimate, the methodology for estimating work activity exposures, the intended dose outcome, and the accuracy of dose rate and man-hour estimates

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

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

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

- Source term control strategy or justifications for not pursuing such exposure reduction initiatives

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

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

- Resolution through the corrective action process of problems identified through post-job reviews and post-outage ALARA report critiques

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

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

The inspector completed 11 of the required 15 samples and 6 of the optional samples.

b. Findings

1. Introduction. The inspector identified a Green finding because the licensee's failures associated with work planning, implementation, and oversight resulted in Radiation Work Permit 05-3530, "Reactor Vessel Head Inspection in Restricted High Radiation Areas," exceeding 5 person-rem and 150 percent of its estimated dose.

Description. The initial dose estimate for Radiation Work Permit 05-3530 was 3.040 person-rem, but the actual dose accrued was 13.614 person-rem. The work duration was planned for 7 days, but actually required 64 days. The licensee attempted to use unproven technology for the inspection work, resulting in many problems and delays. While reviewing the work package, the inspector noted that projected dose estimates for the extended work were poorly documented. The inspectors also reviewed the licensee's root cause analysis and noted that there were problems with the licensee's work planning, implementation, and oversight. The project plan did not address logistical issues, contain success criteria, or use site-operating experience. Neither the vendor contract nor the project plan described the minimum flaw size that is required to be seen during the inspection. An Electric Power Research Institute report stated that the probes developed by the engineering firm failed to meet testing criteria prior to the outage. Project challenges prior to and during the inspection were not adequately presented to the management team. Management oversight was based on trust and resulted in limited intrusiveness.

Analysis. The failure to adequately plan work conducted in restricted high radiation areas was a performance deficiency. This finding was more than minor because it was associated with the occupational radiation safety cornerstone attribute (ALARA planning/estimated dose) and affected the associated cornerstone objective in that it increased the collective dose. The finding involved a failure to adequately plan the head inspection and resulted in unplanned, unintended occupational collective dose for the work activity. When processed through the occupational radiation safety significance determination process, this ALARA finding was found to have no more than very low safety significance because: (1) the finding was related to ALARA planning or work control, but (2) the licensee's 3-year rolling average collective dose was not greater than 135 person-rem. In addition, this finding had crosscutting aspects associated with human performance. The ALARA planners' failure to take proper actions directly caused the finding. The work group failed to address problems that caused the unplanned dose through the corrective action program

Enforcement. No violation of regulatory requirements occurred. The finding was documented in the licensee's corrective action program in Condition Report 200501853. FIN 05000285/2005004-02, Failure to adequately plan work in restricted high radiation areas.

2. <u>Introduction</u>. The inspector identified a Green finding because the licensee's failure to plan emergent work resulted in Radiation Work Permit 05-3519, "SI-220 Valve Replacement in Restricted High Radiation Areas," exceeding 5 person-rem and 150 percent of its estimated dose.

<u>Description</u>. The initial dose estimate for Radiation Work Permit 05-3519 was 2.970 person-rem, but the actual dose accrued was 9.843 person-rem. The original ALARA planning was appropriate; however, problems occurred during the work activity, which required the licensee to perform more work than originally planned. For example, a shield plug that was supposed to be inserted into a severed pipe did not fit properly, so an alternate shielding technique had to be devised. Also, inflatable bladders designed to prevent water from entering new weld areas did not work adequately, so additional welding repair was necessary. There was no indication in the ALARA work package that all emergent work was planned by ALARA personnel so that dose estimates could be formulated and dose saving measures considered. After adjusting the original dose projection to account for those circumstances over which the licensee had no control, the actual dose was approximately 151 percent of estimated dose.

During the discussion of this work activity, the inspectors determined that the problems involving the shield plug and the ineffective bladder were not entered into the licensee's corrective action program and consequently, were never fully analyzed prior to proceeding with the work activity.

<u>Analysis</u>. The failure to plan emergent work conducted in restricted high radiation areas was a performance deficiency. This finding was more than minor because it was associated with the occupational radiation safety cornerstone attribute (ALARA planning/estimated dose) and affected the associated cornerstone objective in that it increased the collective dose. The finding involved a failure to plan and resulted in unplanned, unintended occupational collective dose for the work activity. When processed through the occupational radiation safety significance determination process, this ALARA finding was found to have no more than very low safety significance because: (1) the finding was related to ALARA planning or work control, but (2) the licensee's 3-year rolling average collective dose was not greater than 135 person-rem. In addition, this finding had crosscutting aspects associated with human performance and problem identification and resolution. The ALARA planners' failure to take proper actions directly caused the finding. The work group failed to address problems that caused the unplanned dose through the corrective action program

<u>Enforcement</u>. No violation of regulatory requirements occurred. The finding was documented in the licensee's corrective action program in Condition Reports 200504080 and 200504274. FIN 05000285/2005004-03, Failure to adequately plan emergent work in restricted high radiation areas.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

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.

Section 2OS2 describes a NRC-identified finding that had crosscutting aspects associated with Problem Identification and Resolution. Two problems were identified during the SI-220 valve replacement which were not entered into the licensee's corrective action program. Consequently, the problems were never fully analyzed prior to continuing. The inspector believes that, had these two problems been addressed via the corrective action program, better planning would have been performed prior to continuing with the evolution.

Section 1R15.b describes an URI that had crosscutting aspects related to problem identification and resolution in that although the depressurized nitrogen cylinders were identified and documented on multiple occasions, the condition was not properly documented in the corrective action program nor evaluated for significance until June 28, 2005.

4OA5 Other Activities

1. Pre-operational Testing Of An Independent Spent fuel Storage Installation (ISFSI) (60854)

a. Inspection Scope

On September 12-14, 2005, an inspector from Region IV, accompanied by a technical reviewer from the Office of High Level Waste & Repository Safety (HLWRS) visited the Fort Calhoun Station to observe the vacuum sipping process being used for identifying spent fuel assemblies with cladding defects.

The licensee has scheduled transferring spent fuel from the spent fuel pool to dry storage at the ISFSI beginning in February or March of 2006. Accurate characterization of spent fuel as either intact or damaged is a critical part of the process. The inspector reviewed the process for determining which fuel assemblies met the technical criteria for undamaged fuel. The inspection also provided information needed to effectively evaluate the fuel selection process later this year. No issues were identified.

The reviewer from HLWRS was part of the effort in preparation for performing the review of a license application for the Department of Energy's proposed Yucca Mountain geologic repository. The condition of the fuel cladding is important to both fuel handling operations after receipt at the repository site and the performance of the repository. Therefore, the staff needed to understand the sensitivity of the various methods used to identify fuel assemblies with cladding defects. The reviewer also discussed the failure history of the fuel handling equipment at the plant, including the trolley. The equipment failure experience may help the staff to assess the reliability of equipment to be used for the construction and operation of the repository.

b. Findings

No issues were identified.

4OA6 Meetings

Exit Meeting Summary

On August 26, 2005, the 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.

On September 14, 2005, the inspector conducted a telephonic exit meeting to present the Emergency Preparedness inspection results to Mr. C. Simmons, Supervisor, Emergency Planning, who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On September 14, 2005, the inspector conducted an exit meeting with Mr. G. Cavanugh and Mr. K. Erdman regarding the ISFSI inspection activity.

On October 11, 2005, the resident inspectors presented the results of the quarterly inspection effort to Mr. H. Faulhaber, Senior Division Manager, Nuclear Engineering, and other members of licensee management. The inspectors confirmed that no proprietary information was examined during the inspection period. Licensee management acknowledged the inspection findings.

40A7 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 meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

.1 Title 10 of CFR Part 50, Appendix B. Criterion III, "Design Control" states, in part, that Measures shall be established to assure that the applicable design basis ... are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above. Procedure EM-PM-EX-0203, "Molded Case Circuit Breaker Inspection and Test," did not incorporate the appropriate breaker trip design basis information. Because the data provided was not correct, the pressurizer power operated relief block valve, HCV-151, was inoperable for 17 months. Upon discovery of this condition in March 2005, the instantaneous current set points for both block valves were set to the required valves. This issue was entered into the licensee's corrective action program as Condition Report 200504448. The performance deficiency resulted in a finding that was of very low risk significance (Green). The estimated change in core damage frequency was 3.3×10^{-7} representing the risk related to both internal and external initiators. The change in large-early release frequency was not significantly affected by the performance deficiency.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bannister, Plant Manager
B. Blome, Manager, Planning
G. Cavanaugh, Supervisor, Regulatory Compliance
A. Clark, Manager, Security and Emergency Planning
M. Core, Manager, System Engineering
S. Coufal, ALARA Technician
H. Faulhaber, Senior Division Manager, Nuclear Engineering
M. Frans, Assistant Plant Manager
D. Guinn, Station Licensing Engineer
R. Haug, Manager, Radiation Protection
J. Herman, Manager, Engineering Programs
T. Maine, Radiation Protection Supervisor, ALARA
E. Matzke, Station Licensing Engineer
J. McManis, Manager, Licensing
T. Nellenbach, Operations Manager
T. Pilmaier, Manager, Chemistry
M. Puckett, Previous Radiation Protection Manager
C. Simmons, Supervisor, Emergency Planning
C. Snow, ALARA Technician
D. Spires, Manager, Outage and Work Week
M. Tesar, Division Manager, Nuclear Support Services Division
J. Tillis, Manager, Maintenance
R. Westcott, Manager, Nuclear Projects

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000285/2005004-02	FIN	Failure to adequately plan work in restricted high radiation areas. (Section 2OS2)
05000285/2005004-03	FIN	Failure to adequately plan emergent work in restricted high radiation areas. (Section 2OS2)

Opened

05000285/2005004-01	URI	Inoperable Backup Instrument Air to Condensate Makeup Control Valve. (Section 1R15.b)

LIST OF DOCUMENTS REVIEWED

<u>Section 1R15: Operability Evaluations (71111.15)</u>

AOP-17; Loss of Instrument Air; Revision 7

SO-R-2; Condition Reporting and Corrective Action; Revision 30

EOP-20, "Functional Recovery Procedure," Revision 17

PE-PM-VX-0400, "Valve Maintenance," Revision 5

OP-ST-RC-3002, "Reactor Coolant System (RCS) Category B Valve Exercise Test," Revision 3

EM-RR-VX-0406, "MC2 Diagnostic Test Procedure for Motor Operated Valves," Revision 3

<u>Condition Reports</u>

CR 200503231

CR 200001832

CR 199900364

CR 200503877

CR 200501209

CR 200504583

<u>Other</u>

Root Cause Analysis - Level B; CR 200503231; Loss of Nitrogen Backup for Valve LCV-1190 for an Extended Duration

SDBD-FW-AFW-117; Auxiliary Feedwater; Revision 28

Fort Calhoun PRA Summary Notebook; Revision 7

USAR 9.4; Auxiliary Feedwater System; Revision 14

EA-FC-89-054; Station Blackout Coping Assessment; Revision 4

DCN 0010116; LDV-1190 Backup Gas Source

Turbine Building Operator Logs; January 1 to July 12, 2005

System Training Manual, Volume 37, Reactor Coolant System

Composite Flow Diagram, Reactor Coolant System P&ID, E-23866-210-110, Revision 15

SDBD-RC-128, Reactor Coolant Design Basis Document, Revision 18, Attachment 21

Technical Specification 2.1.6, “Pressurizer and Main Safety Valves”

NRC Information Notice, “Inadequate Verification of Over-current Trip Set points in Metal-Clad, Low Voltage Circuit Breakers”

NRC Information Notice 92-03, “Remote Trip Function Failures in General Electric F-Frame Molded-Case Circuit Breakers”

Root Cause Analysis Report, “Tripping of PORV Block Valve HCV-151 Breaker,” dated 9/7/05

Drawing GE-177B2371, Sheet 379, “Data Sheet for Motor Control Center 4A1,” Revision 26

Vendor Manual for Mag Break® Motor Circuit Protectors

Work Order 00126106-01, “Molded Case Circuit Breaker Functional Test” dated 10/3/03

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents

200502079
200502898
200503230
200503557
200503569
200503745
200504274

Audits and Self-Assessments

Quality Assurance Audit Report No 49/58, “Chemistry Control & Radiation Protection”

Radiation Work Permits (or Radiation exposure permit)

05-2512	Reactor Head Work in HRA’s
05-2531	Bare Metal Inspections and UT Pzr Nozzles
05-3005	ECT of Aux Building Heat Exchangers
05-3512	Reactor Head Work in RHRA’s
05-3519	SI-220 in RHRA’s
05-3523	Root and Drain Valve Replacement
05-3530	RX Vessel & CEDM Seal Housing Inspection in RHRA’s
05-3532	Dose Reduction Activities in HRA’s and RHRA’s

Procedures

RP-AD-300	ALARA Program, Revision 12
RP-201	Radiation Work Permits, Revision 27
RP-301	ALARA Job Reviews, Revision 22
SO-G-101	Radiation Worker Practices, Revision 27

ALARA Committee Minutes

Feb 03, 2005
Feb 15, 2005
Mar 08, 2005
Mar 18, 2005
Mar 26, 2005
Mar 30, 2005
May 19, 2005
May 25, 2005
Jun 23, 2005

Miscellaneous

Fort Calhoun Nuclear Station Dose Reduction Plan 2004 - 2009
2005 FCS ALARA and Radiation Protection Program Goals and Implementation Tools

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
AV	Apparent Violation
CCW	Component Cooling Water
CFR	Code of Federal Regulations
HLWRS	High Level Waste Repository
IA	instrument air
ISFSI	Independent Spent Fuel Storage
LOIA	loss of instrument air
NCV	noncited violation
NRC	Nuclear Regulatory Commission
PORV	Power Operated Relief Valve
RCS	Reactor Coolant System
URI	Unresolved Item