UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

May 15, 2007

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 05000285/2007002

Dear Mr. Ridenoure:

On March 31, 2007, 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 9, 2007, with Mr. David Bannister, Plant Manager, and other members of your staff.

The inspection 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 three self-revealing findings of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violations or significance of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Fort Calhoun Station facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosure, will be available electronically for public inspection in the NRC Public Document

Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jeff Clark, P.E.
Chief, Project Branch E
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2007002
w/Attachment: Supplemental Information

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RIV:RI:DRP/E	SRI:DRP/E	C:DRS/EB1	C:DRS/OB	C:DRS/EB2
LMWilloughby;mjs	JDHanna	WBJones	ATGody	LJSmith
GDR for	***GDR for***	***/RA/***	***TFS for***	***/RA/***
5/1/07	5/1/07	5/2/07	5/3/07	5/3/07

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285
License:	DPR-40
Report:	05000285/2007002
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, 2007
Inspectors:	J. Hanna, Senior Resident Inspector L. Willoughby, Resident Inspector D. Stearns, Health Physicist
Approved By:	Jeff Clark, Chief, Project Branch E Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000285/2007002; 01/01/2007 - 03/31/2007; Fort Calhoun Station, Integrated Resident and Regional Report; Access Control to Radiologically Significant Areas, Identification and Resolution of Problems, Event Follow-up.

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

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A Green self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, occurred when operating procedure OP-3A, "Plant Shutdown," Revision 66, did not contain appropriate guidance to licensed operators to prevent the loss of shutdown cooling when reactor coolant pumps were secured. The procedure did not provide a caution statement similar to one found in other procedures that would have alerted the operators that reduced spray flow exists when running less than four reactor coolant pumps.

  This finding was determined to be greater than minor in that it affected the "Procedure Quality" attribute of the Initiating Events cornerstone. The inspectors attempted to evaluate this finding using Manual Chapter 0609, Appendix G, because the condition occurred during cold shutdown conditions. The reactor had been shut down for 79 days and one third of the fuel was replaced with new fuel bundles. The time to boil was three hours, therefore none of the checklists were applicable. Using Checklist 2 as a bounding evaluation resulted in a Green finding. Since the finding was not suitable for analysis under the significance determination process, regional management and a Senior Reactor Analyst review determined that the finding was of very low safety significance (Green) because there was no affect on the reactor coolant system and no radionuclide release. This finding has been entered into the licensee's corrective action program as Condition Report 200605629. This finding has a crosscutting aspect in the area of human performance associated with resources because procedure OP-3A, "Plant Shutdown, Revision 66" did not contain complete and adequate information for the control of pressurizer spray while transiting to shutdown cooling (4OA3.2).

Cornerstone: Mitigating Systems

- <u>Green</u>. A Green self-revealing noncited violation was identified for the licensee's failure to promptly identify and correct a repetitively inoperable component cooling flow element. The initial failure occurred in 1999 and had failed three times within the past two years. The failure to recognize and fix this condition led to the flow element repeatedly being out of service and unable to perform its function during a potential design basis accident.

  This finding was determined to be greater than minor because the condition had an impact on availability/reliability of the component and thus affected the "Equipment Performance" attribute under the Mitigating Systems cornerstone. The inspectors evaluated this finding using Manual Chapter 0609, Appendix A, and determined that it was of very low safety significance (Green). This conclusion was reached because the finding was not a design or qualification deficiency, the finding did not represent a loss of safety function, was not an actual loss of safety function of a single train for greater than its Technical Specification Allowed Outage time, did not represent an actual loss of safety function for non-Technical Specification equipment, and was not potentially significant due to external events such as flooding, seismic occurrences, etc. This violation was entered into the licensee's corrective action program as Condition Report 200605986 (Section 4OA2.1).

Cornerstone: Occupational Radiation Safety

- <u>Green</u>. The inspectors reviewed two examples of a self-revealing, noncited violation of Technical Specification 5.11.1 in which workers failed to obtain high radiation area radiological briefing before entering the area. The first example occurred on October 25, 2006, when a worker received an electronic alarming dosimeter dose alarm while performing duties as a fire watch on one of the steam generator platforms, which was posted as a high radiation area. The second example occurred on October 29, 2006, when a worker received an electronic alarming dosimeter dose alarm while pulling electrical cable inside the bioshield, which was posted as a high radiation area. For both issues, the licensee restricted access to the radiologically controlled area pending discussion with the individuals and their supervisors. This issue was also included as preshift briefings and management meetings to heighten the awareness of changing radiological conditions and for workers to be more mindful of the radiation work permit requirements.

  This finding is greater than minor because it is associated with one of the cornerstone attributes, exposure/contamination control, and affects the Occupational Radiation Safety cornerstone objective in that the failure to obtain high radiation area access authorization and the associated radiological briefings could have resulted in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an as low as reasonably achievable planning or work control issue; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. This finding also has a crosscutting aspect in the area of human performance work control because neither the individuals nor their supervisors appropriately coordinated work activities and evaluated the impact of changes to work assignments. These issues have been entered into the licensee's corrective action program as Condition Report 200604937 and Condition Report 200605033 (Section 2OS1).

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

The unit began this inspection period in Mode 1 at full rated thermal power and operated at 100 percent until the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignments (71111.04)

.1 Partial Equipment Walkdowns

a. Inspection Scope

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

- February 28, 2006, Emergency Diesel Generator 1 Starting Air System
- March 8, 2006, Motor Driven Auxiliary Feedwater Pump FW-6 and associated safety-related portions of the Auxiliary Feedwater System
- March 14, 2006, High Pressure Safety Injection Pump SI-2C and associated portions of the High Pressure Safety Injection System

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

.2 Complete Equipment Walkdown

a. Inspection Scope

The inspectors: (1) reviewed plant procedures, drawings, the licensee’s USAR, Technical Specifications, and vendor manuals to determine the correct alignment of the Instrument Air system; (2) reviewed outstanding design issues, operator work arounds, and USAR documents to determine if open issues affected the functionality of the Instrument Air system; and (3) verified that the licensee was identifying and resolving equipment alignment problems. Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

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

- January 30, 2007, East Electrical Switchgear Area, Room 56E (Fire Area 36A)
- January 31, 2007, Emergency Core Cooling System Pump Rooms, Rooms 21 and 22 (Fire Areas 1 and 2)
- March 6, 2007, Service and Condensate Tank Area, Room 81 (Fire Area 43)
- March 9, 2007, Electrical Penetration Area - Basement Area, Room 20 (Fire Area 34A)
- March 13, 2007, Component Cooling Heat Exchanger Area, Room 33 (Fire Area 33)
- March 26, 2007, Area in the north end of Turbine Building at 1011' elevation, Turbine Building Elevation 990' through 1036' (Fire Area 46.2)

Documents reviewed by the inspectors included: Standing Order SO-G-28, "Station Fire Plan," Revision 66, Standing Order SO-G-102, "Fire Protection Program," Revision 7, Abnormal Operating Procedure AOP-6, "Fire Emergency," Revision 17, and USAR, Section 9.11, and "Fire Protection Systems."

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario observed on March 13, 2007, involved a safety injection tank nitrogen leak, an over feeding of a steam generator, and a reactor coolant system leak.

Documents reviewed by the inspectors included: Procedure AOP-22, "Reactor Coolant Leak," Revision 28, Procedure EOP-22, "Standard Post Trip Actions," Revision 22, Procedure EOP-03, "Loss of Coolant Accident," Revision 33, and Scenario 84201d, "LOCA," Revision 2.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- March 20, 2007, Inoperable containment sump strainers during Fall 2006 outage
- March 29, 2007, Loss of shutdown cooling on November 27, 2006

Documents reviewed by the inspectors included: Cause Determinations 08040610 and 08120701, as well as Condition Reports 200604163, 200603993, and 200605831.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

.1 Risk Assessments and Management of Risk

a. Inspection Scope

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

- January 18, 2007, Reviewed elevated risk condition with Diesel Generator 1 protected to support a surveillance test while heavy equipment was being rigged over Diesel Generator 1 room to be placed on top of containment

- February 23, 2007, Reviewed elevated risk condition while Auxiliary Feed Water Pump FW-6 and Low Pressure Safety Injection Pump SI-1A was taken out of service for 4160 KV breaker Mechanism Operated Contact Offset rod testing

- March 14, 2007, Reviewed elevated risk condition while Diesel Generator 1 and Low Pressure Safety Injection Pump SI-1A were removed from service for a monthly surveillance test and mechanical seal cleaning

Documents reviewed by the inspectors included: Condition Report 200700351 and the daily risk assessment profiles for the dates listed above.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

.2 Emergent Work Control

a. Inspection Scope

The inspectors: (1) verified that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work-related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the USAR to determine if the licensee identified and corrected risk assessment and emergent work control problems.

- January 25, 2007, Reviewed risk condition that was elevated from baseline when an emergent Blair city water outage occurred to repair a leak in the supply line to the plant

Documents reviewed by the inspectors included the daily risk assessment profiles and the list of compensatory measures for the date listed above.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- February 13, 2007, Containment Ventilation Units VA-3 and VA-7 low component cooling water delta-pressure alarms

- February 23, 2007, Safety-related 4160 KV breaker Mechanism Operated Contact offset rod failure

- March 12, 2007, Normal Range Stack Gas Radiation Monitor Remote Ratemeter RM-062 red alarm light being dimly lit and flickering

- March 19, 2007, Auxiliary Feed Water Pump FW-6 motor end lube oil sight glass oil leak

Documents reviewed by the inspectors included: Condition Reports 200700680, 200700618, 200700620, 200606014, and 200701174.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

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

- January 26, 2007, Replace Filter Regulator Assembly for test procedure IA-HCV-474-FR

- January 26, 2007, Troubleshoot Boric Acid Tank CH-11B level indication transmitter loop L-254

- March 5, 2007, Perform Steam Generator RC-2B Auxiliary Feedwater Inlet Valve limit switch HCV-1108-33A adjustment or minor repair

- March 6, 2007, Install new constants in Spec 200 Feedwater Controllers for Steam Generators RC-2A and RC-2B

- March 20, 2007, Troubleshoot and replace Feedwater Pump FW-4C Recirculation Valve FCV-1151C-20 and replace Feedwater Pump FW-4C Recirculation Control Valve FCW-1151C Pressure Switches PS-1151C and PS-1151F

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- March 1, 2007, Review of surveillance test OP-ST-DG-0002, "Diesel Generator 2 Check," Revision 51

- March 6, 2007, Observed surveillance test OP-ST-VA-0001, "Monthly Hydrogen Purge Test," Revision 9

- March 8, 2007, Observed surveillance test OP-PM-AFW-0004, "Third Auxiliary Feedwater Pump Operability Verification," Revision 29

- March 13 and 14, 2007, Reviewed in service Test OP-ST-RW-3002A, "Raw Water System Category A and B Valve Exercise Test," Revision 10 for HCV-2881A

- March 23, 2007, Observed surveillance test OP-ST-FP-0001D, "Fire Protection System inspection and Test," Revision 16

Documents reviewed by the inspectors included: Fire Impairment 2007-110; Standing Order SO-G-103, "Fire protection Operability and Surveillance Requirements, Revision 20; and the documents listed above.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

For the drills listed below and simulator-based training evolutions contributing to Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) Performance Indicators, the inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in classification, notification, and Protective Action Recommendations (PAR) development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with the guidance of the NEI 99-02, "Voluntary Submission of Performance Indicator Data," acceptance criteria.

- February 27, 2007, Emergency Planning Drill was conducted, which involved a rupture of a waste gas decay tank, a loss of main feedwater, a loss of auxiliary feedwater, a medical emergency, and a loss of once thru core cooling.

Documents reviewed by the inspectors included: Procedure AOP-9, "High Radioactivity," Revision 9, Procedure EOP-20, "Functional Recovery Procedure," Revision 21; Procedure EPIP-OSC-1, "Emergency Plan Implementing Procedure," Revision 41; Procedure EPIP-EOF-7, "Protective Action Guidelines," Revision 18; and Report EP-07-043, "Drill Report for the Feb 27th, 2007 EP Drill," dated March 27, 2007.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

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

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

- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas

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

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

- 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

- 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

The inspectors completed 13 of the required 21 samples.

b. Findings

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

Description. The first example occurred on October 25, 2006, when a worker received a dose alarm on their electronic alarming dosimeter (EAD) when performing duties as a fire watch on the steam generator platform. The individual was initially assigned to the tool room using Radiation Work Permit (RWP) 06-2542, task 4. This task had an EAD dose alarm setting of 10 millirem. The individual was then reassigned, by their supervisor, to one of the steam generator platforms as a fire watch. This task required the individual to use RWP 06-3530, task 3, and required a daily briefing prior to entering the area. The individual stated they knew they needed to switch to the other RWP, but forgot to switch to the appropriate RWP and obtain the access briefing. The RWP task for fire watch duties on the steam generator platform would have changed the EAD dose limit from 10 millirem to 80 millirem. The individual received an EAD dose alarm, immediately exited the area, and reported to radiation protection. The individual's EAD showed an alarm at 10.1 millirem. A restriction was placed on the individual's access pending notification and discussion with their supervision.

The second example occurred on October 29, 2006, when an electrician involved in pulling cable inside containment received a dose alarm on their EAD. The individual was pulling cable outside of the bioshield wall using RWP 06-2527, task 1, which had an EAD dose alarm setting of 10 millirem. The individual stated that he thought he was signed in on task 2, which allowed entry into high radiation areas and had an EAD dose alarm setting of 50 millirem. When the individual went inside the bioshield, a posted high radiation area that requires a radiological briefing, he received an EAD dose alarm. The individual immediately exited the area and contacted radiation protection. The individual's EAD indicated he had received 10.4 millirem. A review of radiological briefing records did not show that the individual had received the required briefing. Exclusion was placed on the individual until Radiation Protection management could discuss this issue with the employee and their supervision.

Analysis. The failure to obtain required high radiation area radiological briefings before entering high radiation areas is a performance deficiency. This finding is greater than minor because it is associated with one of the cornerstone attributes, exposure/contamination control, and affects the Occupational Radiation Safety cornerstone objective in that the failure to obtain high radiation area access authorization and the associated radiological briefings could have resulted in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that this finding was of very low safety significance because it did not involve: (1) an as low as is reasonably achievable (ALARA) planning or work control issue; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. This finding also has a crosscutting aspect in the area of human performance work control because neither the individuals nor their supervisors appropriately coordinated work activities and evaluated the impact of changes to work assignments.

Enforcement. Plant Technical Specifications, Section 5.11.1, states, in part, "in lieu of the 'control device' required by 10 CFR 20.1601(a) and 20.1601(c), each high radiation area, as defined in 10 CFR 20.1601, shall be barricaded and conspicuously posted as an HRA and entrance thereto controlled by a RWP. Any individuals permitted to enter such areas shall be provided with a continuously integrating and alarming radiation monitoring device and may enter after the dose rate levels in the area have been established and personnel are made knowledgeable of them." Contrary to the Technical Specification requirements, workers entered high radiation areas without obtaining the required radiological briefings and were not specifically authorized to enter the area. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as Condition Reports 200604937 and 200605033, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: Noncited Violation (NCV) 05000285/2007002-01, Failure to obtain high radiation area access authorizations and associated radiological briefing.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

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

- Current 3-year rolling average collective exposure, currently in fourth quartile with 169 Rem
- Six 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
- Six work activities of highest exposure significance completed during the last outage
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Intended versus actual work activity doses and the reasons for any inconsistencies
- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling, and engineering groups
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents

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

- Shielding requests and dose/benefit analyses

- Dose rate reduction activities in work planning

- Postjob (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 person-hour estimates

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

- 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, 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 follow-up activities, such as initial problem identification, characterization, and tracking

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

The inspectors completed 22 of the required 29 samples

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Initiating Events

The inspectors sampled submittals for the performance indicators listed below for the period January 1, 2006, through December 31, 2006. The definitions and guidance of Nuclear Engineering Institute (NEI) 99-02, “Regulatory Assessment Indicator Guideline,” Revision 4, were used to verify the licensee’s basis for reporting each data element in order to verify the accuracy of performance indicator data reported during the assessment period.

- IE1 Unplanned Scrams
- IE2 Scrams with a Loss of Normal Heat Removal
- IE3 Unplanned Power Changes

The inspectors completed the three samples in this cornerstone.

Cornerstone: Occupational Radiation Safety

- Occupational Exposure Control Effectiveness

The inspectors reviewed licensee documents from August 1, 2006, through February 26, 2007. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee’s technical specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the performance indicators data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicators definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

The inspectors completed the required sample, one, in this cornerstone.

Cornerstone: Public Radiation Safety

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

The inspectors reviewed licensee documents from August 1, 2006, through February 26, 2007. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicators thresholds and those reported to the NRC. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the performance indicators data. Performance indicators definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

The inspectors completed the required sample, one, in this cornerstone.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Failure to Promptly Identify and Correct an Inoperable Component Cooling Water Flow Element

Introduction. A Green self-revealing noncited violation was identified for the licensee's failure to promptly identify and correct a repetitively inoperable component cooling flow element. The initial failure occurred in 1999 and has failed three times within the past two years. The failure to recognize and fix this condition led to the flow element repeatedly being out of service and unable to perform its function during a potential design basis accident.

Description. On December 14, 2006, flow indicator FI-417 (Flow Indicator for Cooling Water Flow from VA-1B) failed resulting in alarms in the control room. The flow indicator and the associated containment cooler were subsequently declared inoperable and Technical Specification 2.4, "Containment Cooling," was entered. On December 15, 2006, the licensee installed a temporary modification to bypass the signal from the flow indicator and exited the Technical Specification Action Statement.

Containment cooling following a design basis accident (e.g., a Main Steam Line Break or Loss of Coolant Accident) is provided by coolers. The component cooling water system provides the heat removal capacity for this system and FI-417 measures the return flow downstream of one of the coolers. The flow element provides an automatic signal to isolate two containment isolation valves during a Containment Isolation Actuation Signal coincident with low flow through the cooler. The functions of this component are to 1) provide flow indication for containment cooler VA-1B, and 2) automatically isolate a component cooling water leak in containment so as to preserve the integrity of the remainder of the system.

The inspectors began reviewing this condition following an examination of the Condition Reporting system. The inspectors determined that three previous failures of the flow element had occurred on January 1, 1999, April 27, 2005, and December 26, 2005. Further, the inspectors observed that a cause had not been accurately determined for any of the three failures, nor had any (effective) corrective actions been taken. The cause had initially been attributed to flow induced vibration, whereas, the damage is now believed to have been caused by installation of the flow element combined with design deficiencies. The inspectors questioned the licensee about potential causes and the extent of condition to components with a similar design. No other degraded/inoperable components were identified. In addition, the inspectors found that the broken instrument tubes for the flow elements had not been recovered from the system. Through walkdowns of the system and reviews of isometric drawings, the inspectors did not identify any downstream piping or components that might potentially be affected by the loose parts.

Analysis. The inspectors determined that the failure to promptly identify and correct a condition adverse to quality was a performance deficiency. This finding was determined to be greater than minor because the condition had an impact on availability/reliability of the component and thus affected the "Equipment Performance" attribute under the Mitigating Systems cornerstone. The inspectors evaluated this finding using Manual Chapter 0609, Appendix A, and determined that it was of very low safety significance (Green). This conclusion was reached because the finding was not a design or qualification deficiency, the finding did not represent a loss of safety function, was not an actual loss of safety function of a single train for greater than its Technical Specification Allowed Outage time, did not represent an actual loss of safety function for non-Technical Specification equipment, and was not potentially significant due to external events such as flooding, seismic occurrences, etc.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, the licensee did not take prompt corrective actions to correct the inoperable flow indicator FI-417 after identification of the problem on January 1, 1999. This resulted in the flow element being unavailable to perform its safety function on several occasions. Since this failure to take prompt corrective action is of very low safety significance and was documented in the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2007002-02). This violation was entered into the licensee's corrective action program as Condition Report 200605986.

.2 Routine Reviews of Identification and Resolution of Problems

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

.3 Crosscutting Issue Aspects

The inspectors identified one finding with problem identification and resolution crosscutting aspects. As described in Section 4OA2, licensee personnel repeatedly failed to identify and correct failed component cooling water flow elements.

4OA3 Event Followup (71153)

.1 (Closed) LER 05000285/2005002-00, Inoperability of Pressurizer Power Operated Relief Block Valve Due to Human Error

On March 9, 2005, the licensee identified that the circuit breaker providing power to the motor for one of the power operated relief valve (PORV) block valves (HCV-151) had an incorrect instantaneous overcurrent setting. This condition led to intermittent tripping of the breaker during valve operation and the inoperability of the valve for 17 months. This licensee-identified finding involved a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” The enforcement aspects of the violation were addressed in Section 4OA7 of NRC Inspection Report 05000285/2005004. This LER is closed.

.2 (Closed) LER 05000285/2006008-00, Loss of Shutdown Cooling Due to Repressurizing Reactor Coolant System

a. Inspection Scope

The inspectors reviewed control room response to a loss of shutdown cooling event on November 27, 2006. As part of the follow-up, the inspectors reviewed control room logs and the root cause analysis report.

b. Findings

Introduction: A Green self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, occurred when operating procedure OP-3A, “Plant Shutdown,” Revision 66, did not contain appropriate guidance to licensed operators to prevent the loss of shutdown cooling when reactor coolant pumps were secured. The procedure did not provide a caution statement, similar to one found in other procedures that would have alerted the operators that reduced spray flow exists when running less than four reactor coolant pumps.

Description: On November 27, 2006, the licensee was cooling down the reactor coolant system per procedure OP-3A, “Plant Shutdown,” Revision 66, to repair an in-core instrument flange leak that was discovered during the hot hydrostatic test. The reactor coolant system was in a cold shutdown condition with the reactor coolant system pressure at approximately 230 psia and temperature at approximately 120ºF. Two reactor coolant pumps in loop 1 (RC-3A and RC-3B) were running to provide cooling via steam generator RC-2A. All the pressurizer heaters were on and normal spray flow was maximized. The reactor shutdown cooling system was in service with low pressure safety injection Pumps SI-1A and SI-1B running in preparation to transfer to shutdown cooling.

The licensee secured the reactor coolant pumps, which also secured normal pressurizer spray. This caused reactor coolant system pressure to commence rising due to having all the pressurizer heaters on. The system pressure increased to the automatic isolation of shutdown cooling setpoint (250 psia) and shutdown cooling was isolated. The licensee recognized the loss of shutdown cooling and entered abnormal operating procedure AOP-19, “Loss of Shutdown Cooling,” Revision 12. Initially the licensee attempted to lower reactor coolant system pressure using the normal pressurizer spray controller, but realized this had no effect and switched to auxiliary pressurizer spray using the chemical and volume control system. Concurrently, the pressurizer heaters were secured along with the low pressure safety injection pumps. Reactor coolant system pressure was reduced to below 250 psia and shutdown cooling was restored using low pressure safety injection pump SI-1A. The duration that shutdown cooling was lost was twelve minutes. The plant and pressurizer cool-down was recommenced and the leaking in-core instrument flange repaired.

Analysis. The inspectors determined that not providing adequate procedural guidance for the control of pressurizer spray was a performance deficiency. This finding was determined to be greater than minor in that it affected the "Procedure Quality" attribute of the Initiating Events cornerstone. The inspectors attempted to evaluate this finding using Manual Chapter 0609, Appendix G, because the condition occurred during cold shutdown conditions. The reactor had been shut down for 79 days and one third of the fuel was replaced with new fuel bundles. The time to boil was three hours with the reactor coolant system closed, therefore, none of the checklists were applicable. Using Checklist 2 as a bounding evaluation resulted in a green finding since core heat removal was available via reactor coolant pumps and steam generators, inventory control was maintained via the chemical and volume control system along with high pressure safety injection system, both offsite and onsite power sources were available, and containment control was being maintained. Since the finding was not suitable for analysis under the significance determination process, regional management and a Senior Reactor Analyst determined that the finding was of very low safety significance (Green), because there was no effect on the reactor coolant system and no radionuclide release. This finding has a crosscutting aspect in the area of human performance associated with resources because procedure OP-3A, "Plant Shutdown," Revision 66, did not contain complete and adequate information for the control of pressurizer spray while transiting to shutdown cooling.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings. Contrary to the above, on November 27, 2006, the licensee's plant shutdown procedure did not contain appropriate guidance to licensed operators to prevent the loss of shutdown cooling when reactor coolant pumps were secured. Specifically procedure OP-3A, "Plant Shutdown," Revision 66, did not provide a caution statement similar to those found in other procedures that would have alerted the operators that reduced normal spray flow exists when running less than four reactor coolant pumps. This resulted in reactor coolant pressure increasing above the automatic isolation of shutdown cooling setpoint and caused the isolation of shutdown cooling. Because the inadequate procedure resulted in a very low safety significance finding, and it has been entered into the licensee's corrective action program as Condition Report 200605629, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2007002-03).

.3 (Closed) LER 05000285/2007001-00, Inoperability of a Post Accident Main Steam Line Radiation Monitor due to Procedural Error

On February 1, 2007, a station employee identified that all four steam isolation valves for RM-064, Main Steam Line Radiation Monitors, (MS-100, MS-101, MS-102, and MS-103), were closed. The valves are required to be open and function to supply steam from both steam generators to Radiation Monitor RM-064 for post accident radiation monitoring. Immediate compensatory measures were instituted and permanent repairs and postmaintenance testing were completed on February 6, 2007. The licensee determined that this condition was introduced on November 29, 2006, during the Fall 2006 Refueling Outage. The licensee determined the cause to be inadequate procedural guidance in procedure OI-FW-6, "Steam Generator Draining," Revision 38, with contributing causes of human performance in the area of work practices (i.e., procedural compliance). This finding is more than minor in that the Equipment Performance attribute (i.e., availability) of the Mitigating System Cornerstone was affected. The finding was considered to have very low safety significance (Green) using Appendix A of the significance determination process, because the condition did not affect the systems required to function following an accident. This licensee identified violation involved a violation of Technical Specification 2.21. The enforcement aspects of the violation are discussed in Section 4OA7 of this report. This LER is closed.

4OA6 Meetings

Exit Meeting Summary

On March 01, 2007, the inspectors presented the ALARA inspection results to Mr. J. Reinhart and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

The results of the resident inspector activities were presented to Mr. D. Bannister, Plant Manager, and other members of licensee management on April 9, 2007. The inspectors confirmed that proprietary information examined during the inspection period was returned to the licensee. Licensee management acknowledged the inspection findings.

4OA7 Licensee-Identified Violations

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

- Technical Specification 2.21, "Post-Accident Monitoring Instrumentation," required that a Main Steam Line Radiation Monitor be operable. Without the instrumentation being operable, a preplanned alternate method of monitoring is required to be taken and the channel must be restored within seven days. Contrary to the above, the Main Steam Line Radiation Monitor was inoperable for 65 days. This finding only had very low safety significance because the condition did not effect the systems required to function following an accident. This finding was identified in the licensee's corrective action program as Condition Report 200700542 and was reported as LER 05000285/2007001-00.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bannister, Plant Manager
G. Cavanaugh, Supervisor, Regulatory Compliance
L. Cherko, RP Technician, ALARA
M. Core, Manager, System Engineering
S. Coufal, Health Physicist, ALARA
H. Faulhaber, Division manager, Engineering
M. Ferm, Manager, Shift Operations
D. Guinn, Licensing Engineer
R. Haug, Radiation Protection Manager
P. Kellogg, Senior Radiation Protection Technician, ALARA
D. Lakin, Manager, Corrective Action Program
T. Maine, Supervisor, ALARA
E. Matzke, Compliance Engineer
J. McManis, Manager, Licensing
T. Nellenbach, Manager, Operations
J. Reinhart, Site Director
C. Simmons, Supervisor, Emergency Preparedness
J. Tills, Manager, Maintenance
C. Williams, Supervisor, HP Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000285/2007002-01	NCV	Failure to Obtain High Radiation Area Access Authorizations and Associated Radiological Briefing (Section 2OS1)
05000285/2007002-02	NCV	Failure to Promptly Identify and Correct an Inoperable Component Cooling Water flow Element (Section 4OA2)
0500285/2007002-03	NCV	Loss of Shut Down Cooling Due to Inadequate Procedure (Section 4OA3)

Closed

05000285/2005002-00	LER	Inoperability of Pressurizer Power Operated Relief Block Valve Due to Human Error (Section 4OA3)
05000285/2006008-00	LER	Loss of Shutdown Cooling Due to Repressurizing Reactor Coolant System (Section 4OA3)

05000285/2007001-00	LER	Inoperability of a Post Accident Main Steam Line Radiation Monitor due to Procedural Error (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1RO4: Equipment Alignment

Drawing B120F07001 Sh. 1, “Starting Air System Schematic DG-1 (Rm. 63) P&ID,” Revision 34

Drawing B120F14001, “Gauge Panel Bay Assembly,” Revision 7

Procedure OI-DG-1, “Diesel Generator No. 1,” Checklist OI-DG-1-CL-A, “DG-1 Starting Air,” Revision 42

Drawing 11405-M-253 Sh. 4, "Flow Diagram Steam Generator Feedwater and Blowdown P&ID," Revision 34

Drawing 11405-M-254 Sh. COV, "Composite Flow Diagram Condensate P&ID," Revision 47

Procedure OI-AFW-1, "Auxiliary Feedwater Actuation System Normal Operation," Checklist OI-AFW-1CL-A, "Auxiliary Feedwater," Revision 65

Procedure OI-AFW-4, "Auxiliary Feedwater Startup and System Operation," Revision 62

Drawing E-23866-210-130 Sh. 3, "Safety Injection and Containment Spray System Flow Diagram P&ID," Revision 16

Procedure OI-SI-1, "Safety Injection - Normal Operation," Checklist OI-SI-1-CL-A, "HP Safety Injection System," Revision 101

Drawing 11405-M-10 Sh 4, "Auxiliary Coolant Component Cooling System Flow Diagram P&ID," Revision 8

Procedure OI-CC-1, 'Component Cooling System Normal Operation," Checklist OI-CC-1-CL-A, "Component Cooling," Revision 58

Procedure OI-CA-1, “Compressed Air Normal Operation,” Revision 56

System Training Manual, Volume 43, “Service and Instrument Air System,” Revision 8

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

Corrective Action Documents

CR-200501675, CR-200604123, CR-200604938, CR-200604937, CR-200605033, CR-200605908

Radiation Work Permits

06-3536; Reactor Head Replacement in Restricted High Radiation Areas
06-2336; Rapid Refueling Project
06-3532; NSSSRP Preparation and Rigging
06-3530; Steam Generator Cutout and Weld-in

Procedures

SO-G-101, Radiation Work Practices, Revision 30
RP-204, Radiological Area Controls, Revision 46
RP-306, Hot Spot Identification and Tracking, Revision 17

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents

CR-200603601, CR-200603627, CR-200603736, CR-200603848, CR-200604266, CR-200604525, CR-200604623, CR-200605908, CR-200606060, CR-200606084, CR-200606099, CR-200606104, CR-200700068, CR-200700340, CR-200700630, CR-200700763

Audits and Self-Assessments

Self-Assessment 07-09; 2006 ALARA Package Review
Self-Assessment 07-08; Effectiveness of the RP Count Room and Contamination Monitors
06-QUA-043; Quality Department Surveillance Report, Radiation Work Permits - ALARA

Radiation Work Permits

06-3512; Reactor Head Disassembly
06-3520; Valve Maintenance
06-2336; Rapid Refueling Project
06-3532; NSSSRP Preparation and Rigging
06-3530; Steam Generator Cutout and Weld-in
06-2540; Replacement of Containment Sump Strainers

Procedures

RP-302, ALARA Planning/RWP Development and Control, Revision 26
RP-AD-300, ALARA Program, Revision 13
RP-305, ALARA Suggestion Program, Revision 5
RP-608, Dose Calculations from Contamination, Revision 12
RP-650, Internal Dosimetry Program, Revision 10
RP-655, In-Vitro Bioassay Sampling, Revision 4
RP-656, Bioassay Calculations, Revision 5
RP-907, Radiological Analysis, Revision 3
RP-306, Hot Spot Identification and Tracking, Revision 17

Miscellaneous

Fort Calhoun Dose Reduction Plan, 2006-2011, dated 2/22/07
Radiological Analysis, Determination of Personnel Contamination Monitor Efficiency Correction Factors

Section 4OA2: Identification and Resolution of Problems (71152)

Control Room Operating Logs from December 14 to December 15, 2006
Design Drawing 02405 - Ellison Instrument Division, "741 to 744 Annubar Flow Elements"
Isometric Drawing 12772, "Raw Water Auxiliary Building," Revision 7
Isometric Drawing 19694, "Raw Water Auxiliary Building," Revision 9
Isometric Drawing 12727, "Raw Water Auxiliary Building," Revision 9
Isometric Drawing 35377, "Raw Water Turbine Building," Revision 9
Fort Calhoun Drawing 11405-M-40, "Auxiliary Coolant Component Cooling System," Revision 36
Fort Calhoun Drawing 11405-M-100, "Raw Water Flow Diagram," Revision 91
Fort Calhoun Drawing 11405-M-257, "Flow Diagram Circulating Water," Revision 27

Condition Reports

200700997	200700721	200605986	200600304	200505889	200505848
200505837	200502621	200401747	200501375	200401571	200200199
200101679	199900885	199900001			

Section 4OA5: Other Activities (71111.05T)

Procedures

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

Drawings

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

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

Calculations

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

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

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

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

FC 05814, “UFHA Combustible Loading,” Revision 9

Condition Reports

200204316	200503731	200503738	200503750	200503979	200504001
200504006	200504203	200601090			

Miscellaneous

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

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

Work Orders

00208529-01	00219055-01	00261046-01	00263608-01	00263610-01
00264582-01	00264582-02			

LIST OF ACRONYMS

ALARA	as low as reasonable achievable
CFR	*Code of Federal Regulations*
DEP	drill exercise performance
EAD	electronic alarm dosimeter
ERO	emergency response organization
PORV	power operated relief valve
RWP	radiation work permit
SSC	structure system component
USAR	Updated Safety Analysis Report