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

February 4, 2004

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 05000285/2003006

Dear Mr. Ridenoure:

On December 31, 2003, 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 January 9, 2004, with Mr. Dave Bannister, Manager, Fort Calhoun Station, 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.

Based on the results of this inspection, the NRC identified five findings that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC also determined that there were violations associated with each of these findings. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violation or significance of these 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, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Fort Calhoun Station facility.

In accordance with 10 CFR 2.790 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 (PARS) 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/ by WCWalker Acting for**

Kriss M. Kennedy, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2003006
w/attachment: Supplemental Information

cc w/enclosure:
John B. Herman, Manager
Nuclear Licensing
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

Richard P. Clemens, Division Manager
Nuclear Assessments
Fort Calhoun Station
P.O. Box 550
Fort Calhoun, NE 68023-0550

David J. Bannister, Manager - Fort Calhoun Station
Omaha Public Power District
Fort Calhoun Station FC-1-1 Plant
P.O. Box 550
Fort Calhoun, NE 68023-0550

James R. Curtiss
Winston & Strawn
1400 L. Street, N.W.
Washington, DC 20005-3502

Chairman
Washington County Board of Supervisors
P.O. Box 466
Blair, NE 68008

Sue Semerena, Section Administrator
Nebraska Health and Human Services System
Division of Public Health Assurance
Consumer Services Section
301 Centennial Mall, South
P.O. Box 95007
Lincoln, NE 68509-5007

Daniel K. McGhee
Bureau of Radiological Health
Iowa Department of Public Health
401 SW 7th Street, Suite D
Des Moines, IA 50309

Chief Technological Services Branch
National Preparedness Division
Department of Homeland Security
Emergency Preparedness & Response Directorate
FEMA Region VII
2323 Grand Boulevard, Suite 900
Kansas City, MO 64108-2670

Electronic distribution by RIV:
Regional Administrator **(BSM1)**
DRP Director **(ATH)**
DRS Director **(DDC)**
Senior Resident Inspector **(JGK)**
Branch Chief, DRP/C **(KMK)**
Senior Project Engineer, DRP/C **(WCW)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
Dan Merzke, Pilot Plant Program **(DXM2)**
**RidsNrrDipmLipb**
Debby Jackson, OEDO RIV Coordinator **(DAJ1)**
FCS Site Secretary **(NJC and VLH)**
W. A. Maier, RSLO **(WAM)**
Dale Thatcher **(DFT)**

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| LMWilloughby | JGKramer | TWPruett | CSMarschall | KMKennedy |
| ***T - WCWalker*** | ***T - WCWalker*** | ***/RA/*** | ***/RA/*** | ***WCWalker for*** |
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**U.S. NUCLEAR REGULATORY COMMISSION**

REGION IV

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| Docket: | 50-285 |
| License: | DPR-40 |
| Report: | 05000285/2003006 |
| Licensee: | Omaha Public Power District |
| Facility: | Fort Calhoun Station |
| Location: | Fort Calhoun Station  FC-2-4 Adm.<br>P.O. Box 399, Hwy. 75 - North of Fort Calhoun<br>Fort Calhoun, Nebraska |
| Dates: | September 21 through December 31, 2003 |
| Inspectors: | J. Kramer, Senior Resident Inspector<br>L. Willoughby, Resident Inspector<br>J. Adams, Reactor Inspector<br>L. Ellershaw, Senior Reactor Inspector<br>R. Lantz, Senior Emergency Preparedness Inspector<br>M. Maley, Reactor Operations Engineer, Nuclear Reactor Regulation<br>G. Miller, Reactor Inspector<br>N. O'Keefe, Senior Reactor Inspector<br>L. Ricketson, P.E., Senior Health Physicist |
| Approved By: | Kriss M. Kennedy, Chief, Project Branch C<br>Division of Reactor Projects |

Enclosure

**CONTENTS**

SUMMARY OF FINDINGS . . . . . . . . . . 1

REACTOR SAFETY . . . . . . . . . . 1
1RST Postmaintenance and Surveillance Testing . . . . . . . . . . 1
1R01 Adverse Weather Protection . . . . . . . . . . 2
1R04 Equipment Alignments . . . . . . . . . . 3
1R05 Fire Protection . . . . . . . . . . 3
1R07 Heat Sink Performance . . . . . . . . . . 4
1R08 Inservice Inspection Activities . . . . . . . . . . 5
1R11 Licensed Operator Requalification . . . . . . . . . . 6
1R12 Maintenance Rule Implementation . . . . . . . . . . 7
1R13 Maintenance Risk Assessments and Emergent Work Evaluation . . . . . . . . . . 8
1R14 Operator Performance During Nonroutine Evolutions and Events . . . . . . . . . . 9
1R15 Operability Evaluations . . . . . . . . . . 11
1R16 Operator Workarounds . . . . . . . . . . 11
1R20 Refueling and Other Outage Activities . . . . . . . . . . 12
1R23 Temporary Plant Modifications . . . . . . . . . . 14
1EP4 Emergency Action Level and Emergency Plan Changes . . . . . . . . . . 14

RADIATION SAFETY . . . . . . . . . . 15
2OS1 Access Control to Radiologically Significant Areas . . . . . . . . . . 15
2OS2 As Low as is Reasonably Achievable (ALARA) Planning and Controls . . . . . . 20

OTHER ACTIVITIES . . . . . . . . . . 22
4OA1 Performance Indicator Verification . . . . . . . . . . 22
4OA2 Problem Identification and Resolution . . . . . . . . . . 23
4OA3 Event Followup . . . . . . . . . . 25
4OA5 Other . . . . . . . . . . 28
4OA6 Meetings . . . . . . . . . . 29
4OA7 Licensee-Identified Violations . . . . . . . . . . 29

SUPPLEMENTAL INFORMATION . . . . . . . . . . A-1
KEY POINTS OF CONTACT . . . . . . . . . . A-1
LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED . . . . . . . . . . A-1
LIST OF DOCUMENTS REVIEWED . . . . . . . . . . A-2

## SUMMARY OF FINDINGS

IR 05000285/2003006; 09/21/2003 - 12/31/2003; Fort Calhoun Station, Integrated Resident and Regional Report; Maintenance Rule Implementation, Nonroutine Evolutions and Events, Refueling Activities, Access Control, Event Followup

The report covered a 3-month period of inspection by Resident and Regional office inspectors. Five Green noncited violations of significance were identified. The significance of the findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>NRC-Identified Findings and Self-Revealing Findings</u>

Cornerstone: Initiating Events

- Green. A noncited violation was identified for the failure of the licensee to have an adequate procedure for properly venting the reactor vessel head as required by 10 CFR Part 50, Appendix B, Criterion V. This resulted in a compressed bubble being formed in the reactor vessel and a false and nonconservative indication of reactor vessel level.

  This finding was more than minor since it is associated with the procedure quality attribute of the cornerstone. The finding was characterized as having very low safety significance because the core heat removal, inventory control, electrical power, containment control, and reactivity control support systems were available (Section 1R14.1).

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion V, was identified as a result of the failure of the licensee to establish a procedure for the verification of the reactor coolant system parameters during a plant heatup. This resulted in the control room staff being unaware of a recently added surveillance requirement to, in part, monitor reactor coolant system parameters during a heatup.

  This finding was more than minor since it is associated with the procedure quality attribute of the cornerstone. The finding was characterized as having very low safety significance because the plant heatup was performed using decay heat with the pressurizer vented; therefore, the chance of exceeding pressure and temperature limits was minimal. This finding also had crosscutting aspects associated with human performance (Section 1R20.1).

Cornerstone: Mitigating Systems

- Green. A noncited violation was identified as a result of the licensee failing to demonstrate that performance of Raw Water Pump AC-10B was being effectively controlled through the performance of appropriate preventive maintenance. Following a failure of the pump, the licensee failed to consider placing the system under 10 CFR 50.65(a)(1) for establishing goals and monitoring against the goals.

This finding was more than minor since it met the example of a “not minor finding” in Inspection Manual Chapter 0612, Appendix E. The finding was characterized as having very low safety significance because the maintenance rule aspect of the finding did not cause an actual loss of safety function of the system nor did it cause a component to be inoperable. This finding also had crosscutting aspects associated with human performance (Section 1R12).

Cornerstone: Barrier Integrity

- Green. A noncited violation was identified as a result of the failure of the spent fuel handling machine operator to follow the procedure for transferring fuel in the spent fuel pool as required by Technical Specification 5.8.1.a. This failure resulted in the dropping of a fuel assembly in the spent fuel pool.

  This finding was more than minor since it is associated with the fuel cladding human performance attribute of the cornerstone. The finding was characterized as having very low safety significance because there was no damage to fuel pins or breach of the spent fuel storage pool liner. This finding also had crosscutting aspects associated with human performance (Section 4OA3.1).

Cornerstone: Occupational Radiation Safety

- Green. A noncited violation, with three examples, was identified as a result of the licensee’s failure to barricade, conspicuously post, and lock or guard restricted high radiation areas to prevent unauthorized entry as required by Technical Specifications 5.11.1 and 5.11.2.

  Example One. A radiation protection technician walked through a door to Steam Generator Bay B on the 994-foot elevation of the containment building and left the door unguarded and open with the posting not conspicuous. General area dose rates were as high as 1500 millirem per hour in the bay.

  Example Two. The ladder leading to Steam Generator Bay A from the steam generator platform was locked with a sheet metal gate, but the gate was flanked on the side by rails which were approximately 3 feet high. This would have allowed an individual to bypass the gate by simply stepping over the railing. General area dose rates were as high as 4000 millirem per hour in the bay.

  Example Three. A permanent ladder leading into the reactor cavity from the south side was controlled by locking the ladder climbing rails at the top of the ladder. An individual could either step around the ladder barrier or go underneath it and enter the reactor cavity. Additionally, on the north side of the cavity, scaffolding was erected to house a set of temporary stairs into the cavity. An individual could bypass the locked door by climbing on the outside of the scaffolding and down into the reactor cavity using a ladder-like structure which was part of the scaffolding. General area dose rates were as high as 5000 millirem per hour in the reactor cavity.

This finding was more than minor because inadequate controls of high radiation areas affect the licensee's ability to ensure adequate protection of worker health and safety from exposure to radiation. Because the finding involved the potential for workers to receive significant, unplanned, unintended doses as a result of conditions, the finding was evaluated using Appendix C of the Occupational Radiation Safety Significance Determination Process. The finding was characterized as having very low safety significance because a substantial potential for overexposure did not exist. Example One of this finding also had crosscutting aspects associated with human performance (Section 2SO1).

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 action tracking numbers (condition report numbers) are listed in Section 4OA7 of this report.

## REPORT DETAILS

Summary of Plant Status

The unit began this inspection period in Mode 5 in a refueling outage. Core off-load commenced on September 22, 2003, and was completed on September 26. Core re-load commenced on October 16 and was completed 2 days later. On October 25 operators commenced a heatup to Mode 3. On October 30 operators performed a reactor startup and the unit was synchronized to the grid on October 31. On November 7 the unit achieved 100 percent power. On November 10 operators reduced power to 92 percent due to the failure of a second circulating water pump. On November 17 the unit returned to 100 percent power and operated at that power level throughout the remainder of this inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1RST Postmaintenance and Surveillance Testing (71111.ST)

.1 Postmaintenance Tests

a. Inspection Scope

The inspectors observed and/or reviewed four postmaintenance tests 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 00119987-01, to refurbish auxiliary feedwater inlet Valve HCV-1108A, including Procedure PE-RR-VX-0414S, “Inspection and Repair of Safety Related Fisher HSC Control Valves,” Revision 3, on October 28, 2003
- Work Order 00153272-01, to inspect Charging Pump CH-1C speed reducer and couplings on December 16, 2003
- Work Order 00155656-01, to clean, inspect, and megger the Containment Spray Pump SI-3A motor, check the space heater operation and winding resistance, obtain a lube oil sample, and change lube oil in the inboard and outboard motor bearings on December 18, 2003
- Work Order 00162293-01, to troubleshoot and repair Diesel Generator 1 Turbo Oil Circulating Pump Motor, LO-40-1-M, on December 19, 2003

b. Findings

No findings of significance were identified.

.2 Surveillance Tests

a. Inspection Scope

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

- OP-ST-CCW-3022, “AC-3C Component Cooling Water Pump Inservice Test,” Revision 14
- QC-ST-MX-3003, “Visual Inspection (VT-2) of Piping Areas of Limited Access,” Revision 2, Attachment 9.1, “Inspection Under Reactor Vessel”
- OP-ST-ESF-0002, “Diesel Generator NO.1 and NO. 2 Auto Operation,” Revision 27, for Diesel Generator 1
- IC-ST-AE-3833, “As Found Type C Local Leakrate Test of M-HCV-383-3,” Revision 11

b. Findings

No findings of significance were identified.

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed Procedure OI-EW-1, “Extreme Weather,” Revision 8, for responding to extreme weather, specifically cold weather preparations. The inspectors evaluated the design features and implementation of the procedure to protect the raw water system and auxiliary feedwater system from the effects of adverse weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Equipment Walkdowns

a. Inspection Scope

The inspectors performed two partial walkdowns 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:

- Raw water pumps during an outage of Raw Water Pump AC-10A, Raw Water Component Cooling Water Heat Exchanger AC-1C, and Raw Water Strainer AC-12A on November 18, 2003

- Diesel Generator 1 fuel oil system while Diesel Generator 2 was inoperable for monthly testing on December 10, 2003

b. Findings

No findings of significance were identified.

.2 Complete System Walkdowns

a. Inspection Scope

The inspectors conducted a detailed review of the alignment and condition of the low pressure safety injection system. The inspectors reviewed open work orders and condition reports associated with the system. The inspectors performed a walkdown of accessible portions of the system. During the walkdown, inspectors verified that the system was properly aligned in accordance with piping and instrumentation Drawing E-23866-210-130 and Procedure OI-SI-1, “Safety Injection - Normal Operation,” Revision 70.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Fire Inspection Tours

a. Inspection Scope

The inspectors performed seven routine fire inspection tours 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 1 - Safety Injection and Containment Spray Pump Area (Room 21)
- Fire Area 2 - Safety Injection and Containment Spray Pump Area (Room 22)
- Fire Area 20.7 - Railroad Siding Area (Room 25)
- Fire Area 31 - Intake Structure, Raw Water Pump area
- Fire Area 33 - Component Cooling Heat Exchanger Area (Room 18)
- Fire Area 34C - Group 1 MCC Area (Room 57E)
- Fire Area 35A - Diesel Generator (Room 1)

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation

a. Inspection Scope

The inspectors observed a plant fire drill on December 11, 2003, and evaluated the readiness of licensee personnel to prevent and fight fires. The inspectors placed an inspection emphasis on proper donning of fire gear, use of a self-contained breathing apparatus, entry into the fire area, fire brigade leader's directions, simulated use of firefighting equipment, and communications. The inspectors discussed any observations with the evaluator following the drill scenario.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the performance of Component Cooling Water Heat Exchanger AC-1C by reviewing one performance test. Test acceptance criteria and results were reviewed to ensure differences between testing conditions, and design conditions were considered. In addition, the inspectors reviewed the test results against pre-established engineered acceptance criteria and Condition Report 200305760.

b. Findings

No findings of significance were identified.

## 1R08 Inservice Inspection Activities (71111.08)

### .1 Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections

#### a. Inspection Scope

The inspectors observed the ultrasonic system calibration and ultrasonic examinations of 11 reactor vessel head-to-flange studs (RPV-G1-S-20 through RPV-G1-S-30). During the review of these examinations, the inspectors verified that the correct nondestructive examination procedure was used, examinations and conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also reviewed the documentation to determine if the indications revealed by the examinations were compared against the American Society of Mechanical Engineers (ASME) Code specified acceptance standards and whether indications were appropriately dispositioned. The nondestructive examination certifications of those personnel observed performing examinations or identified during review of completed examination packages were reviewed by the inspectors.

The inspectors reviewed modification activities associated with the addition of ASME Code Section III, Class 2 piping to allow for reactor coolant system thermal expansion of Reactor Coolant System Loops 1A and 2A between Regenerative Heat Exchanger CH-6 and reactor coolant system loop injection Valves HCV-247 and HCV-248 performed under Construction Work Order CWO-03-0046, Revision 6. The inspectors reviewed the welding procedure specification (WPS 801) and its associated procedure qualification records to assure that Section IX requirements of the ASME Code were met.

Additionally, the inspectors reviewed the weld data forms and the final liquid penetrant examination and visual examination reports for 12 of the ASME Code Section III, Class 2 reactor coolant system thermal expansion piping welds: F-23C-1, F-23C-2, F-23C-3, F-23C-4, F-23C-6, F-23C-7, F-23C-8, F-23C-10, F8-A1, F8-A2, F8-A3, and F8-A4.

#### b. Findings

No findings of significance were identified.

### .2 Steam Generator Tube Inspection Activities

#### a. Inspection Scope

The inspectors reviewed and verified that the steam generator tube eddy current examination scope and expansion criteria met the Technical Specification requirements, Electric Power Research Institute Guidelines, and commitments made to the NRC. In conjunction with this review, the inspectors confirmed that known areas of potential degradation were included in the scope of inspection.

The inspectors observed portions of eddy current examinations performed on the following locations in Steam Generators A and B: hot-leg side between 3 inches above the top of the tubesheet to 7 inches below the top of the tubesheet and cold-leg side dent locations. During these examinations, the inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered to, and (4) probe travel speed was in accordance with procedural requirements.

Due to an unplanned refueling outage delay, steam generator tube eddy current examinations were stopped for approximately one day; thus, there was no opportunity to observe implementation of any potential repairs (e.g., plugging operations) or in-situ pressure testing.

The inspectors reviewed Report SG-SGDA-02-026, “Fort Calhoun Station Steam Generator Operational Assessment For Cycle 21, Spring 2002,” which evaluated and summarized the results of the 2002 refueling outage steam generator inspection and testing activities. The purpose was to demonstrate that the structural and leakage integrity criteria were expected to be maintained throughout Cycle 21 (which ended with the initiation of the current refueling outage [03RFO]). In addition, the inspectors reviewed Report SG-SGDA-03-16, “Fort Calhoun Station Steam Generator Degradation Assessment,” the purpose of which was to provide a comprehensive review and overall plan for detection and assessment of degradation to be addressed during Refueling Outage 03RFO. Further, the report provided predictions as to the type and extent of degradation expected to be found. At the time of the inspectors’ review, the licensee had completed approximately 50 percent of the scheduled eddy current examinations, the results of which appeared to be on track with the predictions.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

The inspectors performed one licensed operator requalification observation. On December 17, 2003, the inspectors observed licensed operator requalification training activities, including the licensed operators’ performance and the evaluators’ critique. The inspectors compared performance in the simulator with performance observed in the control room during this inspection period. The focus of the inspection was on high-risk licensed operator actions, 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 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 the licensee personnel. The inspectors reviewed Condition Reports 200203511, 200301794, and 200303410 and the maintenance rule functional scoping data sheet for the raw water pumps. The following three maintenance rule items were reviewed:

- Control Room Air Conditioning Units VA-46A and VA-46B
- Diesel Generator 1 Turbo Oil Circulating Pump Motor LO-40-1-M
- Raw Water Pump AC-10B

b. Findings

Introduction. A Green noncited violation was identified as a result of the licensee failing to demonstrate that performance of Raw Water Pump AC-10B was being effectively controlled through the performance of appropriate preventive maintenance. Following a failure of the pump, the licensee failed to consider placing the system under 10 CFR 50.65(a)(1) for establishing goals and monitoring against the goals.

Description. The inspectors performed a review of Condition Report 200301794 that documented the failure of Raw Water Pump AC-10B during a quarterly surveillance test on May 15, 2003. The inspectors questioned a maintenance rule engineer about the rationale for not including the failure in the maintenance rule program. The engineer indicated that the pump failure was incorrectly classified as not being a maintenance rule functional failure and initiated Condition Report 200303410 to address the finding.

Analysis. The inspectors evaluated the safety significance of the finding. Upon review of Inspection Manual Chapter 0612, Appendix E, Example 1.f, the inspectors determined that this finding was more than minor since the failure of the raw water pump caused the system to be placed in Maintenance Rule Category a(1). This finding was characterized under the significance determination process as having a very low safety significance because the maintenance rule aspect of the finding did not cause an actual loss of safety function of the system nor did it cause a component to be inoperable.

This finding had crosscutting aspects associated with human performance. The failure of licensee personnel to correctly identify the failure of the pump as a maintenance rule functional failure was due to a lack of adequate review of the component failure and directly contributed to the finding.

Enforcement. Section 50.65(a)(1) of 10 CFR requires, in part, that holders of an operating license shall monitor the performance or condition of SSCs within the scope of the monitoring program as defined in 10 CFR 50.65(b) against licensee-established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. 10 CFR 50.65 (a)(2) states, in part, that monitoring as specified in 10 CFR 50.65 (a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. Contrary to the above, the licensee failed to demonstrate that performance of the raw water system was being effectively controlled through the performance of appropriate preventive maintenance in that a maintenance preventable failure of Raw Water Pump AC-10B occurred during a quarterly surveillance test on May 15, 2003. Following the failure, the licensee failed to consider placing the raw water pump under 10 CFR 50.65(a)(1) for establishing goals and monitoring against the goals. Although the licensee recognized that the pump failed its surveillance test and corrected the deficiency, they did not recognize that the failure was a maintenance rule functional failure. This violation of 10 CFR 50.65(a)(2) is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 285/2003006-01). This violation was entered into the licensee's corrective action program as Condition Report 200303410.

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

### a. Inspection Scope

The inspectors reviewed two licensee risk assessments 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:

- Outage of Motor-Driven Auxiliary Feedwater Pump FW-6, Charging Pump CH-1B, and Circulating Water Pump CW-1B on November 5, 2003

- Outage of Diesel Generator 1, and Circulating Water Pumps CW-1B and CW-1C on November 12, 2003

b. Findings

No findings of significance were identified.

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

.1 Reactor Coolant System Level Transient

a. Inspection Scope

The inspectors observed the control room operators' performance during an indicated lowering of reactor vessel water level and entry into Procedure AOP-19, "Loss of Shutdown Cooling," Revision 8. The inspectors discussed the event with operations and engineering personnel. The inspectors reviewed Procedure OI-RC-2A, "RCS Fill and Drain Operations," Revision 38, and the evaluation of the level transient documented in Condition Report 200303706.

b. Findings

Introduction. A Green noncited violation was identified as a result of the failure of the licensee to ensure that an adequate procedure existed for properly venting the reactor vessel head as required by 10 CFR Part 50, Appendix B, Criterion V.

Description. On September 17, 2003, with the unit shutdown for a refueling outage and reactor coolant temperature approximately 100°F, operators observed a lowering of the indicated level in the reactor coolant system. Within the next few minutes, the indicated level lowered from 1012'6" to 1010'3". Operators entered Procedure AOP-19 as a result of the level decrease. After a few minutes, with 80 gallons per minute charging flow and with letdown isolated, the reactor vessel level started to rise. Approximately 20 minutes later, operators stabilized the reactor vessel level at the pre-event level of 1012'6".

The operators initiated a search for the potential loss of inventory by performing numerous containment and auxiliary building walkdowns. The operators were unable to identify the source of leaking water. After further investigation, the licensee determined that, when craft personnel performed work on a heated junction thermocouple, the craft personnel noted that air was coming out of the thermocouple when a clamp was removed. This work activity created a vent path in the reactor vessel and a rapid depressurization. The depressurization caused a change in indicated level (actual vessel level did not lower) and resulted in the indicated level accurately reflecting the true reactor vessel level. The licensee concluded that no inventory was lost. The operators were previously unaware that the approved procedural method being used to vent the head, prior to the event, was inadequate and resulted in a false and nonconservative reactor vessel level.

The licensee performed an evaluation to determine the actual reactor vessel level following the midloop evolution. The licensee determined that, when midloop conditions were exited on September 16, the operators secured filling the reactor vessel when the actual reactor vessel level was approximately 1008’6" and the indicated level was 1012’6". The top of the hot leg was 1007’8". The inspectors independently calculated the lowest actual vessel level and concluded licensee’s calculations were accurate. Over the next day, operators periodically added water (a total of 1500 gallons) to the reactor coolant system as the head slowly vented. The operators did not recognize there was a reactor vessel inventory and level indication problem, although 1500 gallons of water were added with a nearly constant reactor coolant temperature.

With the actual reactor vessel level at 1008'6" and assuming a complete loss of shutdown cooling (decay heat removal), the time-to-boil was 28 minutes and time until the core would uncover was 259 minutes. The licensee had maintained the shutdown operations protection plan prior to and during the event. The shutdown operations protection plan ensured the availability of redundant equipment in independent trains for the mitigation of accidents. Specifically, a redundant means of inventory injection were available throughout the event.

Analysis. The inspectors evaluated the safety significance of the finding. This finding affected the initiating events cornerstone and was considered more than minor because the procedure did not establish an adequate vent of the reactor vessel head. Based on the results of Phase 1 of the significance determination process evaluation performed using Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determinations Process,” the inspectors determined that the finding did not require a quantitative assessment. Therefore, the finding was characterized as having very low safety significance (Green).

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions or procedures of a type appropriate to the circumstances. Procedure OI-RC-2A, “RCS Fill and Drain Operations,” Revision 38, provided documented instructions for filling the reactor coolant system and ensuring the reactor vessel head is vented. Contrary to the above, the licensee failed to have documented instructions of a type appropriate to the circumstances to ensure the reactor vessel head was properly vented when filling the reactor coolant system when exiting midloop conditions. This failure resulted in nonconservative indicated reactor vessel level. This violation of 10 CFR Part 50, Appendix B, Criterion V, is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 285/2003006-02). This violation was entered into the licensee’s corrective action program as Condition Report 200303706.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- Containment Pump Outlet Strainer SI-12A, openings in the inner screen mesh were larger than expected (Condition Report 200304391)

- Capability of component cooling water to provide design flow to supported components based on analytical results rather than actual test data (Condition Report 200305471)

- Diesel Generator 1 Turbo Lube Oil Circulating Pump, LO-40-1, motor seized (Condition Report 200305638)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Selected Operator Workaround

a. Inspection Scope

The inspectors performed a selected review of an operator workaround due to Temporary Modification EC 33591 that replaced the position indication for Control Element Assembly 20 with the position indication of Control Element Assembly 21 in the secondary control element assembly position indication system. The inspectors discussed human reliability in responding to an initiating event with Operations supervision, specifically the effect of the operator workaround on the operator's ability to implement abnormal or emergency operating procedures. The inspectors discussed the planned corrective actions for the deficiency with Operations supervision.

b. Findings

No findings of significance were identified.

.2 Cumulative Effects of Operator Workarounds

a. Inspection Scope

The inspectors performed a review of the operator workaround and the control room deficiency lists. The inspectors focused on the cumulative effects of the workarounds on the reliability and availability of mitigating systems. The inspectors reviewed Procedure OPD-4-17, “Control Room Deficiencies and Operator Work Arounds,” Revision 10, that described the programs for handling workarounds and deficiencies. The inspectors discussed the programs and planned corrective actions for the deficiencies with Operations supervision.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

.1 Verification of Parameters during Reactor Coolant System Heatup

a. Inspection Scope

The inspectors observed the operators perform a heatup of the reactor coolant system and questioned the operation’s crew on the verification and documentation of the heatup. The inspectors reviewed Technical Specifications, Procedure OP-2A, “Plant Startup,” Revision 48, and Condition Reports 200304812, 200304849, and 200304891.

b. Findings

Introduction. A Green noncited violation of 10 CFR Part 50, Appendix B, Criterion V, was identified as a result of the failure of the licensee to establish a procedure for the verification of the reactor coolant system parameters during a plant heatup.

Description. On October 21, 2003, the inspectors observed portions of the reactor coolant system heatup from 90°F to 130°F. The inspectors questioned an operator about how the heatup was being monitored. The operator indicated that reactor coolant system temperature and heatup rate were being monitored on a strip chart. The inspectors noted that the pressurizer was vented and, therefore, RCS pressure could not increase.

Technical Specification Surveillance Requirement, Table 3-5, Number 23, requires operators, in part, to monitor reactor coolant system parameters during a heatup. The inspectors searched for a procedure or instructions that implemented the Technical Specification requirement and could not locate one. The inspectors then questioned the Shift Manager on how operators ensured that the Technical Specification requirements were met. The licensee initiated Condition Report 200304812 to evaluate the issue.

The licensee determined that Technical Specification Amendment 221, effective on September 5, 2003, was never properly implemented in Procedure OP-2A, “Plant Startup.” The licensee revised Procedure OP-2A to include the Technical Specification requirement.

The inspectors determined that the licensee was performing a plant heatup, an activity affecting quality, without an approved documented procedure to verify reactor coolant system parameters. In addition, the inspectors determined that the control room staff was unaware of the recently added surveillance requirement to, in part, monitor reactor coolant system parameters during a heatup. However, the inspectors concluded that the operators were appropriately monitoring the plant conditions during the heatup and indirectly met the intent of the Technical Specification Surveillance Requirement.

Analysis. The inspectors evaluated the safety significance of the finding. This finding affected the barrier integrity cornerstone and was considered more than minor because a procedure did not exist to monitor reactor coolant system parameters during a plant heatup. Based on the results of Phase 1 of the significance determination process evaluation performed using Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determinations Process,” the inspectors determined that the finding did not require a quantitative assessment. In addition, the heatup was performed using decay heat and the pressurizer was vented; therefore, the chance of exceeding pressure and temperature limits was minimal and the finding was characterized as having very low safety significance.

This finding had crosscutting aspects associated with human performance. The failure of licensee personnel to create documented instructions to perform a Technical Specification surveillance directly contributed to the finding.

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions or procedures of a type appropriate to the circumstances. Contrary to the above, on October 21, 2003, during a reactor coolant system heatup from approximately 90°F to 130°F (an activity affecting quality), procedural guidance did not exist to implement the Technical Specification requirement to monitor reactor coolant system parameters during a heatup. The licensee had implemented a Technical Specification Surveillance Requirement on September 5, 2003, that, in part, required the verification of the reactor coolant system temperature, pressure, and heatup rate every 30 minutes. However, the licensee failed to create a procedure to implement the new requirement. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 285/2003006-03). This violation was entered into the licensee’s corrective action program as Condition Report 200304812.

.2 Other Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the licensee's refueling outage shutdown risk assessment to verify that the licensee appropriately considered risk in planning and scheduling the outage activities. The inspectors observed and reviewed the lowering of reactor coolant system water level to midloop conditions, core fuel off-load and core on-load, shutdown maintenance activities, plant heatup, and power ascension. The inspectors performed several containment tours and verified containment cleanliness prior to closure. The inspectors verified that the activities were performed in accordance with approved procedures and Technical Specification requirements. Periodically, the inspectors evaluated plant conditions to verify that safety systems were properly aligned and that maintenance activities were controlled in accordance with the outage risk control plan.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Modification EC 33589 that replaced the position indication for Control Element Assembly 41 with the position indication of Control Element Assembly 40 in the secondary control element assembly position indication system. In addition, the inspectors attended the plant review committee that approved the temporary modification and reviewed the associated 10 CFR 50.59 screening. The inspectors verified that the modification had no adverse impact on the safety function 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 conducted an in-office review of Revision 17 to the Definitions and Abbreviations Section of the Radiological Emergency Response Plan, submitted on September 22, 2003. The inspectors compared the revision to the previous revision, the requirements of 10 CFR 50.54(q) and 10 CFR Part 50, Appendix E, to determine if the revisions decreased the effectiveness of the emergency plan. This revision was administrative in nature to change format and add consistency with other plan sections. These changes are subject to future inspections to ensure that the impact of the changes continues to meet NRC regulations. The inspectors completed the one inspection requirement 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

To review and assess the licensee’s performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, and high radiation areas, the inspectors interviewed supervisors, radiation workers, and radiation protection personnel involved in high dose rate and high exposure jobs during the 2003 refueling outage. The inspectors discussed changes to the access control program with the Radiation Protection Manager. The inspectors also conducted plant walkdowns within the radiological controlled area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Area postings, radiation work permits (RWPs), radiological surveys, and other controls for airborne radioactivity areas, radiation areas, and high radiation areas
- High radiation area key control
- One example in which the internal dose assessment exceeded 50 millirem committed effective dose equivalent
- Setting, use, and response of electronic personnel dosimeter alarms
- Prejob briefings prior to foreign object search and retrieval from Steam Generator B (RWP 03-3532) and prior to draining the reactor upper cavity (RWP 03-3534)
- Dosimetry placement when work involved a significant dose gradient (steam generator work, under-vessel inspections)
- Controls involved with the storage of highly radioactive items in the spent fuel pool

- Audits and self-assessments involving high radiation area controls and staff performance
- Summary of corrective action documents written since the last inspection and selected documents relating to high radiation area incidents, radiation protection technician and radiation worker errors, and repetitive, significant individual deficiencies.

Performance indicator reviews associated with Occupational Exposure Control effectiveness are documented in Section 4OA1 of this report. No licensee event reports or special reports were required in this inspectable area since the previous inspection. The inspectors completed all 21 of the required samples.

b. Findings

Introduction. A Green noncited violation, with three examples, was identified as a result of the licensee's failure to barricade, conspicuously post, and lock or guard a restricted high radiation area to prevent unauthorized entry as required by Technical Specifications 5.11.1 and 5.11.2.

Example One

Description. On October 2, 2003, as the licensee prepared to replace Safety Injection Valve SI-208 in Steam Generator Bay B, the inspectors watched television images of the work activity from the remote monitoring station. Prior to this work, the licensee had measured dose rates within Steam Generator Bay B of 1500 millirems per hour at 30 centimeters from the source of radiation. Therefore, the door at the entrance to the area was posted as a restricted high radiation area, and a radiation protection technician was assigned to guard the door to prevent unauthorized entry into the area. (The licensee's Technical Specifications define an area in which the intensity of radiation is greater than 1000 millirems per hour but less than 500 rads per hour as a restricted high radiation area.)

At one point during the work activity, the inspectors noted that the radiation protection technician assigned to prevent unauthorized access walked through the doorway, into the steam generator bay, and out of sight. This left the door open and the area unguarded. When radiation protection personnel in the remote monitoring station failed to notice the open door and absent door guard, the inspectors identified the situation to them. They contacted the radiation protection technician by radio and advised the technician to return to guard the door. The licensee's review of the occurrence confirmed that the radiation protection technician had not maintained the doorway within sight after entering the bay.

Analysis. This finding was greater than minor because inadequate controls of restricted high radiation areas affect the licensee's ability to ensure adequate protection of worker health and safety from exposure to radiation. Because the finding involved the potential for workers to receive significant, unplanned, unintended doses as a result of conditions contrary to Technical Specification requirements, the inspectors used the Occupational Radiation Safety Significance Determination Process described in Manual Chapter 0609, Appendix C, to analyze the significance of the finding. The inspectors determined that a substantial potential for overexposure did not exist; therefore, the finding had very low significance.

This finding had crosscutting aspects associated with human performance. When licensee personnel left the door to Steam Generator Bay B unlocked, open, unattended, and without being conspicuously posted, their actions directly contributed to the finding.

Enforcement. Section 20.1003 of 10 CFR defines a high radiation area as an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (100 millirem) in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.

Section 20.1601 of 10 CFR requires controls of high radiation areas. Section 20.1601(c) of 10 CFR allows the licensee to apply to the NRC for approval of alternative methods for controlling access to high radiation areas. Technical Specification 5.11 implements the licensee’s alternate methods. Technical Specification 5.11.1 requires that each high radiation area be barricaded and conspicuously posted as a high radiation area. Technical Specification 5.11.2 requires that each high radiation area in which the intensity of radiation is greater than 1000 millirems per hour (a restricted high radiation area) but less than 500 rads per hour have locked doors to prevent unauthorized entry into such an area.

The licensee violated these requirements when the door to Steam Generator Bay B was left open, unlocked, and unattended. Because the door was open, it did not serve as a barricade. Additionally, with the door open, the area posting on the door was not conspicuous to approaching personnel. The radiation protection technician was inside the room with the door out of sight and could not have prevented inadvertent entry into the restricted high radiation area. This is the first example of a violation of Technical Specifications 5.11.1 and 5.11.2 and is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 285/2003006-04). This violation was entered into the licensee’s corrective action program as Condition Report 200304335.

Example Two

Description. On October 28, 2003, while touring the reactor containment building, the inspectors observed that the ladder leading to Steam Generator Bay A from the steam generator platform was barricaded and posted as a restricted high radiation area. Licensee survey records confirmed that dose rates beyond the barricade were as high as 4000 millirems per hour at 30 centimeters from the source of radiation. The inspectors noted that the ladder into the area was controlled with a locked sheet metal gate. However, the gate was flanked on the side by rails which were approximately 3 feet high. This would have allowed an individual to bypass the gate by simply stepping over the railing on either side. Therefore, the inspectors concluded that the licensee's control of the restricted high radiation area was inadequate to prevent unauthorized entry.

Analysis. This finding was greater than minor because inadequate controls of restricted high radiation areas affect the licensee's ability to ensure adequate protection of worker health and safety from exposure to radiation. Because the finding involved the potential for workers to receive significant, unplanned, unintended doses as a result of conditions contrary to Technical Specification requirements, the inspectors used the Occupational Radiation Safety Significance Determination Process described in Manual Chapter 0609, Appendix C, to analyze the significance of the finding. The inspectors determined that a substantial potential for overexposure did not exist; therefore, the finding had very low significance.

Enforcement. Section 20.1003 of 10 CFR defines a high radiation area as an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (100 millirem) in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.

Section 20.1601 of 10 CFR requires controls of high radiation areas. Section 20.1601(c) of 10 CFR allows the licensee to apply to the NRC for approval of alternative methods for controlling access to high radiation areas. Technical Specification 5.11 implements the licensee's alternate methods. Technical Specification 5.11.1 requires that each high radiation area be barricaded and conspicuously posted as a high radiation area. Technical Specification 5.11.2 requires that each high radiation area in which the intensity of radiation is greater than 1000 millirems per hour (a restricted high radiation area) but less than 500 rads per hour have locked doors to prevent unauthorized entry into such an area.

The licensee violated these requirements when the ladder into the Steam Generator Bay A area was controlled with a locked sheet metal gate, but the gate was flanked on the side by rails which were approximately 3 feet high. This would have allowed an individual to bypass the gate by simply stepping over the railing on either side and, therefore, would not have prevented unauthorized entry into the area. This is the second example of a violation of Technical Specifications 5.11.1 and 5.11.2 and is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 285/2003006-04). This violation was entered into the licensee's corrective action program as Condition Report 200305066.

Example Three

Description. On October 28, 2003, the inspectors observed that the reactor cavity was posted as a restricted high radiation area. Licensee survey records confirmed that dose rates within the reactor cavity were as high as 5000 millirems per hour at 30 centimeters from the source of radiation. The inspectors noted that a permanent ladder leading into the reactor cavity from the south side of the cavity was controlled by locking the ladder climbing rails at the top of the ladder. Additionally, on the north side of the cavity, scaffolding was erected to house a set of temporary stairs into the cavity. A locked door was installed, in the horizontal plane, over the temporary stairs to prevent unauthorized use of the stairs. The inspectors concluded that the licensee's control of the restricted high radiation area within the reactor cavity was inadequate to prevent unauthorized access. On the south side an individual could either step around the ladder barrier or go underneath it and enter the reactor cavity. On the north side an individual could bypass the locked door by climbing on the outside of the scaffolding and down into the reactor cavity using a ladder-like structure which was part of the scaffolding.

Analysis. This finding was greater than minor because inadequate controls of restricted high radiation areas affect the licensee's ability to ensure adequate protection of worker health and safety from exposure to radiation. Because the finding involved the potential for workers to receive significant, unplanned, unintended doses as a result of conditions contrary to Technical Specification requirements, the inspectors used the Occupational Radiation Safety Significance Determination Process described in Manual Chapter 0609, Appendix C, to analyze the significance of the finding. The inspectors determined that a substantial potential for overexposure did not exist; therefore, the finding had very low significance.

Enforcement. Section 20.1003 of 10 CFR defines a high radiation area as an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem (100 millirem) in 1 hour at 30 centimeters from the radiation source or 30 centimeters from any surface that the radiation penetrates.

Section 20.1601 of 10 CFR requires controls of high radiation areas. Section 20.1601(c) of 10 CFR allows the licensee to apply to the NRC for approval of alternative methods for controlling access to high radiation areas. Technical Specification 5.11 implements the licensee's alternate methods. Technical Specification 5.11.1 requires that each high radiation area be barricaded and conspicuously posted as a high radiation area. Technical Specification 5.11.2 requires that each high radiation area in which the intensity of radiation is greater than 1000 millirems per hour (a restricted high radiation area) but less than 500 rads per hour have locked doors to prevent unauthorized entry into such an area.

The licensee violated these requirements when controlling access to the reactor cavity. On the south side of the cavity an individual could either step around the ladder barrier or go underneath it and enter the reactor cavity. On the north side an individual could bypass the locked door by climbing on the outside of the scaffolding and down into the reactor cavity using a ladder-like structure which was part of the scaffolding. In both cases, the barrier would not have prevented unauthorized entry into the area. This is the third example of a violation of Technical Specifications 5.11.1 and 5.11.2 and is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 285/2003006-04). This violation is in the licensee's corrective action program as Condition Report 200305066.

2OS2 As Low as is Reasonably Achievable (ALARA) Planning and Controls (71121.02)

a. Inspection Scope

This area was inspected to assess 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 Specification 5.8.1 as criteria for determining compliance. The inspectors interviewed licensee personnel and reviewed:

- Current 3-year rolling average collective exposure
- Site-specific trends in collective exposures, plant historical data, and source-term measurements
- Site-specific ALARA procedures
- Three 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 RWP documents
- Person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements
- 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 man-hour estimates

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

- Exposure tracking system

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

- Workers use of the low dose waiting areas (reviewed September 29 through October 3, 2003)

- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner (reviewed September 29 through October 3, 2003)

- 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

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas (reviewed September 29 through October 3, 2003)

- 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 postjob reviews and postoutage ALARA report critiques

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

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

The inspectors completed 25 samples. Additional samples were completed during inspections documented in NRC Inspection Reports 05000285/2002002 and 05000285/2002005. All 29 samples were completed during the 2002-2003 inspection period.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Cornerstones

a. Inspection Scope

The inspectors reviewed the licensee's performance indicator data to verify its accuracy and completeness for the following two indicators:

- MS1 Emergency ac Power System Unavailability
- MS3 Heat Removal System Unavailability

The inspectors reviewed the performance indicator data for the last quarter of 2002 and the first three quarters of 2003. The inspectors reviewed NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," and licensee operating logs. The inspectors discussed the status of the performance indicators and compilation of data with licensee personnel.

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed corrective action program records involving restricted high radiation areas (as defined in Technical Specification 5.11.2), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned exposure occurrences (as defined in NEI 99-02) for the past 12 months to confirm that these occurrences were properly recorded as performance indicators. Radiologically controlled area entries with exposures greater than 100 millirems within the past 12 months were reviewed, and selected examples were examined to determine whether they were within the dose projections of the governing RWPs. No radiation worker received committed effective dose equivalents of more than 100 millirems, so a review of whole body counts or dose estimates was unwarranted. Where applicable, the inspectors reviewed the summation of unintended deep dose equivalent and committed effective dose equivalent to verify that the total effective dose equivalent did not surpass the performance indicator threshold without being reported. The inspectors completed one of the required samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed radiological effluent release program corrective action records and annual effluent release reports documented during the past four quarters (no licensee event records were submitted) to determine if any doses resulting from effluent releases exceeded the performance indicator thresholds (as defined in NEI 99-02). The inspectors completed one of the required samples.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Cross-Reference to Human Performance Findings Documented Elsewhere

Section 1R12 describes the failure of licensee personnel to perform an adequate review of a component failure and then to correctly identify the maintenance rule failure classification of Raw Water Pump AC-10B and establish performance monitoring goals as required by 10 CFR 50.65(a)(1).

Section 1R20.1 describes the failure of the licensee to ensure that a procedure existed for the verification of the reactor coolant system parameters during a plant heatup as required by 10 CFR Part 50, Appendix B, Criterion V. This failure resulted in the control room staff being unaware of a recently added surveillance requirement to, in part, monitor reactor coolant system parameters during a heatup.

Section 2OS1 Example One describes the failure of the licensee to barricade, conspicuously post, and lock or guard a restricted high radiation area to prevent unauthorized entry as required by Technical Specifications 5.11.1 and 5.11.2. Specifically, a radiation protection technician walked through a doorway into a steam generator bay and left the door open and the area unguarded. In addition, radiation protection personnel at the remote monitoring station failed to notice the open door and absent door guard.

Section 4OA3.1 describes the failure of the spent fuel handling machine operator to follow the procedure for transferring fuel in the spent fuel pool as required by Technical Specification 5.8.1.a. This failure resulted in the dropping of a fuel assembly in the spent fuel pool.

Section 4OA3.2 (licensee identified finding) describes that licensee personnel had incorrectly determined that “inaccessible” because of radiological dose considerations was equivalent to “inaccessible” as defined in the code and therefore did not perform the VT-2 examination of the bottom portion of the reactor vessel as required by Technical Specification 3.3.1.a and Section XI of the ASME Boiler and Pressure Vessel Code.

.2 Selected Issue Followup Inspection

a. Inspection Scope

The inspectors selected 11 condition reports for detailed review (199902690, 200000060, 200000251, 200100155, 200100256, 200100292, 200103752, 200200987, 200200181, 200202442, and 200300658). The condition reports were associated with connection of temporary loads to nonload shed motor control centers via welding receptacles or spare breakers. The reports were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized.

b. Findings and Observations

The licensee’s initial evaluation of the problem was narrow in scope and did not account for spare breakers that could have temporary loads attached. Ultimately, the licensee expanded the scope and proceduralized the use of the receptacles. No findings of significance were identified.

.3 Inservice Inspection Activities

a. Inspection Scope

The inspectors reviewed inservice inspection-related condition reports issued during the past 2 years and verified that the licensee identified, evaluated, corrected, and trended problems. In this effort, the inspectors evaluated the effectiveness of the licensee’s corrective action process, including the adequacy of the technical resolutions.

b. Findings and Observations

No findings of significance were identified.

.4 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution processes relating to high radiation area incidents and radiation protection technician and radiation worker errors.

b. Findings and Observations

No findings of significance were identified.

.5 ALARA Planning and Controls

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution processes regarding exposure tracking, higher than planned exposure levels, and radiation worker practices.

b. Findings and Observations

While reviewing corrective action documents associated with this area on December 10, 2003, the inspectors identified an example in which an action assignment was closed before being completed. Condition Report 200203529, Action Item 1, stated, "Revise Maintenance planning process such that the ALARA group has sufficient detail to provide accurate dose estimates for maintenance work activities performed in the RCA [radiological, controlled area]. Document this process in an applicable guidance document (e.g. MD-AS-0003, SO-M-101, etc)." The inspectors found that the action item was closed on May 30, 2003, without revising the maintenance planning process. No reference to corrective actions in another condition report was provided.

Additionally, the inspectors noted on December 11, 2003, that Condition Report 200304005 stated in the "Response Basis" that an action item was written to make process changes to ensure RWP and ALARA planning were included in the work order planning process. The statement was dated November 18, 2003. However, no action item was written and the condition report was closed December 2, 2003. No reference to corrective actions in another condition report was provided.

To address these items, the licensee initiated Condition Report 200305634 to document ineffective corrective actions and Condition Report 200305664 to document an adverse trend in the licensee's ability to integrate ALARA planning into station work management processes

4OA3 Event Followup (71153)

.1 Dropped Fuel Assembly

a. Inspection Scope

The inspectors reviewed the circumstances and recovery associated with the dropping a fuel assembly in the spent fuel pool on September 23, 2003. The inspectors' review included the procedure requirements for handling fuel in the spent fuel pool, the method used for stabilizing the fuel assembly, the procedure used to recover the fuel assembly, and the licensee's root cause investigation into the event. The inspectors conducted interviews with selected plant personnel and attended plant review committee meetings that discussed stabilization methods, recovery methods, and the corrective actions associated with preventing the recurrence of dropping another fuel assembly. The inspectors observed the stabilization of the fuel assembly and the final placement in the spent fuel pool.

b. Findings

Introduction. A Green noncited violation was identified as a result of the failure of the spent fuel handling machine operator to follow the procedure for transferring fuel in the spent fuel pool as required by Technical Specification 5.8.1.a. This failure resulted in the dropping of a fuel assembly in the spent fuel pool.

Description. On September 23, 2003, at 12:32 p.m. with the core fuel off-load in progress, a fuel assembly was dropped from the spent fuel handling machine while being moved in the spent fuel pool. The assembly dropped approximately 2 feet and came to rest against the spent fuel pool wall. The bottom of the assembly straddled four fuel storage cells and the top was leaning against the spent fuel pool wall.

The spent fuel handling machine operator stopped machine movement and notified the control room. All fuel movement was stopped and the spent fuel pool area was evacuated. Reactor cavity purification was secured and Fuel Transfer Tube Isolation Valve FH-1 was closed. Additional immediate actions were the evacuations of the containment and auxiliary buildings. A Notification of an Unusual Event for plant conditions warranting increased awareness by plant staff or government authorities (EAL 11.4) was declared at 3:20 p.m.

The licensee then developed plans for the stabilization of the fuel assembly and the recovery of the fuel assembly. After plant review committee approval, the licensee installed a fuel stabilization tool to prevent movement while the assembly was being recovered. The fuel assembly was then grappled using a special tool and placed in a storage location in the spent fuel pool. The Notice of Unusual Event was exited at 4:54 p.m. on September 24.

Prior to resuming core off-load, the licensee conducted an investigation to determine why the fuel assembly was dropped. The investigation ruled out mechanical or material failure of the long tool that was used in the movement of fuel assemblies. The investigation concluded that a human performance error caused the assembly to drop while being moved in the spent fuel pool. The licensee then developed corrective actions to prevent a recurrence of this error. After crew briefings and additional training, core off-load recommenced on September 25 and was completed September 26, 2003.

Analysis. The inspectors evaluated the safety significance of the finding. This finding affected the barriers cornerstone and was considered more than minor because the operator did not perform the procedure steps correctly, resulting in the long tool grapple not being latched in the desired position. The finding was characterized under the significance determination process as having very low safety significance (Green) because there was no damage to fuel pins or breach of the spent fuel storage pool liner.

This finding had crosscutting aspects associated with human performance. The failure of the operator on the spent fuel handling machine to follow the procedure for movement of fuel in the spent fuel pool directly contributed to the finding.

Enforcement. Technical Specification 5.8.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Appendix A, requires, in part, written procedures for refueling and core alterations. Procedure OI-FH-1, “Fuel Handling Equipment Operations,” Revision 53, step 9, in part, provides procedure requirements for transferring spent fuel from the upender to the spent fuel pool. Contrary to the above, on September 23, 2003, the operator on the spent fuel handling machine failed to follow the procedure for transferring spent fuel from the upender to the spent fuel pool. This failure resulted in the dropping of a fuel assembly in the spent fuel pool. This violation of Technical Specification 5.8.1.a is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (NCV 285/2003006-05). This violation was entered into the licensee’s corrective action program as Condition Report 200303986.

.2 (Closed) Licensee Event Report 050000285/2003001-00: Noncompliance with Technical Specification Surveillance Requirement for inspection of the lower portion of the reactor vessel

In March 2003, during a boric acid program evaluation, the licensee determined that a VT-2 examination of the bottom portion of the reactor vessel had not been accomplished in accordance with Technical Specification 3.3.1.a and Section XI of the ASME Boiler and Pressure Vessel Code. Licensee personnel had incorrectly determined that “inaccessible” because of radiological dose considerations was equivalent to “inaccessible” as defined in the code and, therefore, had not been performing the required examinations. On September 28, 2003, the licensee satisfactorily performed the required examinations of the bottom of the reactor vessel and did not identify any boric acid deposits. This finding was greater than minor since the failure to implement a Technical Specification-required examination of the bottom of the reactor vessel could become a more significant safety concern if left uncorrected. The finding was characterized under the significance determination process as having very low safety significance (Green) because subsequent examination of the bottom portion of the reactor vessel did not reveal any degradation of the integrity of the reactor coolant system. This licensee-identified finding involved a violation of Technical Specification 3.3.1.a. The enforcement aspects of the violation are discussed in Section 4OA7. This licensee event report is closed.

4OA5 Other

.1 Temporary Instruction 2515/150: Reactor Pressure Vessel Head Penetration Nozzles (NRC Order EA-03-009)

a. Inspection Scope

The inspectors observed and reviewed licensee activities associated with NRC Order EA-03-009 that established interim inspection requirements for reactor pressure vessel heads at pressurized water reactors, issued on February 11, 2003.

b. Findings and Observations

Being ranked a moderate susceptible plant, the licensee performed a 100 percent bare metal visual inspection of the reactor pressure vessel head that included a 360 degree visual inspection around each head penetration nozzle. The ranking was based on plant-specific data used in the calculation of effective degradation years.

The inspection was performed using a contractor developed procedure that was approved by the licensee. The procedure was a combination work agreement and inspection plan. The procedure was qualified on a mockup of the reactor pressure vessel head at the contractor's facility.

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

The licensee's quality control personnel involved with the inspection were VT-2 qualified. They received additional training at the contractor's facility where a mockup of the Fort Calhoun Station reactor pressure vessel head was located. Personnel received training on the operation and limitations of the robotic device. Qualification of the procedure that was used for the examination was accomplished at this time.

The licensee did not identify any boric acid deposits during their inspection. However, the licensee observed boric acid stains in some locations on the reactor pressure vessel head and on some nozzles that were associated with past flange leakage from above. The vessel head contained small debris, dust of light crystals, and a few foreign objects, such as an Allen wrench and small mechanical fasteners. The debris and light crystals were easily scattered with air. No deficiencies were identified that required repairs.

4OA6 Meetings

Exit Meeting Summary

The results of the inservice inspection activities were presented to Mr. R. Ridenoure, Vice President, Nuclear, and other members of licensee management on September 26, 2003. The licensee's management acknowledged the inspection findings and stated that none of the material examined during the inspection was considered proprietary.

The results of the access control to radiological significant areas inspection were presented to Mr. R. Ridenoure, Vice President, Nuclear, and other members of licensee management on October 3, 2003. The licensee's management acknowledged the inspection findings and stated that none of the material examined during the inspection was considered proprietary.

The results of the emergency plan changes inspection were presented to Mr. C. Simmons, Supervisor, Emergency Planning, during a phone conversation on December 3, 2003. The supervisor acknowledged the findings presented and stated that none of the material examined during the inspection was considered proprietary.

The results of the ALARA planning and controls inspection were presented to Mr. M. Franz, Assistant Plant Manager, and other members of licensee management on December 12, 2003. The licensee's management acknowledged the inspection findings and stated that none of the material examined during the inspection was considered proprietary.

The results of the resident inspectors' activities were presented to Mr. D. Bannister, Manager, Fort Calhoun Station, and other members of licensee management on January 9, 2004. The licensee's management acknowledged the inspection findings and noted that some of the material examined during the inspection was considered proprietary. The inspectors indicated that, although examined, no proprietary information was documented in the inspection report.

4OA7 Licensee-Identified Violations

The following findings of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as noncited violations.

- Technical Specification 5.11.1 allows entry into high radiation areas only after individuals have been made knowledgeable of the dose rates in the area. However, on September 16, 2003, an electrician entered a restricted high radiation area on the Steam Generator B platform without having been made knowledgeable of the dose rates in the area. This finding was documented in

Condition Report 200303643. The finding a had very low safety significance because it did not involve a substantial potential for overexposure.

- Technical Specification 5.8.1 requires written procedures be established, implemented, and maintained covering applicable procedures in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Procedures for access control to radiation areas and the use of an RWP system are required. Standing Order SO-G-101, “Radiation Worker Practices,” Revision 22, requires workers to adhere to the requirements of RWPs. However, the licensee identified three examples in which workers failed to follow the requirements of their respective RWPs and entered into areas and radiological conditions that they were not allowed to enter. On January 30, 2003, an individual entered a high radiation area in Room 7 while using RWP 03-1012. This finding was documented in Condition Report 200300325. On September 16, 2003, an individual entered a high radiation area in Steam Generator Bay A while using RWP 03-1543. This finding was documented in Condition Report 200303664. Also on September 16, 2003, an individual entered a high radiation area near the refueling equipment while using RWP 03-1502. This finding was documented in Condition Report 200303672. In each of these three cases, the respective RWP did not allow entry into a high radiation areas. These findings had a very low safety significance because they did not involve a substantial potential for overexposure.

- Technical Specification 3.3.1.a requires, in part, that in-service inspection of ASME Code Class 1 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, as required by 10 CFR Part 50, Section 50.55(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a (g)(6)(i). Section XI of the ASME Boiler and Pressure Vessel Code requires, in part, that a VT-2 examination be performed on the lower portion of the reactor vessel. Contrary to the above, the licensee failed to perform an examination on the lower portion of the reactor vessel and did not have specific written relief granted by the Commission. This violation of Technical Specification 3.3.1.a was documented in the licensee’s corrective action program as Condition Report 200300772. This finding had a very low safety significance because there was no actual degradation of the reactor vessel.

ATTACHMENT: SUPPLEMENTAL INFORMATION

# SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee Personnel

D. Bannister, Manager, Fort Calhoun Station
R. Clemens, Division Manager, Nuclear Assessments
M. Core, Manager, System Engineering
M. Frans, Assistant Plant Manager
R. Haug, Manager, Chemistry
J. Herman, Manager, Nuclear Licensing
R. Phelps, Division Manager, Nuclear Engineering
M. Puckett, Manager, Radiation Protection
R. Ridenoure, Vice President, Nuclear
H. Sefick, Manager, Security and Emergency Planning

## LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 05000285/2003006-01 | NCV | Failure to monitor performance against established goals as required by the Maintenance Rule (Section 1R12) |
| 05000285/2003006-02 | NCV | Inadequate procedure for properly venting the reactor vessel head (Section 1R14.1) |
| 05000285/2003006-03 | NCV | Lack of procedure for the verification of the reactor coolant system parameters during a plant heatup (Section 1R20.1) |
| 05000285/2003006-04 | NCV | Failure to barricade, post, and lock a restricted high radiation area with dose rates greater than 1000 millirems per hour (Section 2OS1) |
| 05000285/2003006-05 | NCV | Failure to follow the procedure for transferring fuel in the spent fuel pool (Section 4OA3.1) |

Closed

| | | |
|---|---|---|
| 05000285/2003001-00 | LER | Failure to inspect the lower portion of the reactor vessel (Section 4OA3.2) |

## LIST OF DOCUMENTS REVIEWED

**Section 1R08: Inservice Inspection Activities**

Work Orders

CWO 03-0046, "Install New Piping and Supports to Allow for Thermal Expansion of RCS Loops 1A and 2A Between the Regenerative Heat Exchanger (CH-6) and Each of the RCS Loop Injection Valves (HCV-247 & 248)."

Procedures

OPPD-UT-89-11, "Ultrasonic Examination of Studs/Bolts Greater Than 2" in Diameter," Revision 1

OPPD-MT-89-1, "Magnetic Particle Examination of Welds and Bolting," Revision 2

QCP-310, "Liquid Penetrant Examination (Solvent Removable)," Revision 12

QCP-300, "Visual Weld Inspection," Revision 10

Welding Procedure Qualifications Records

PQR 801-1
PQR 801-4
PQR Specification #3 dated November 19, 1973
PQR Specification #3 dated May 2, 1987
PQR 331, Revision 1

Test Reports

Ultrasonic Examinations: FCS-UT-1

Liquid Penetrant Examinations: QCIRs 20030551, 20030598, 20030614, 20030661, and 20030672

Visual Examinations: QCIRs 20030661 and 20030672

Miscellaneous

ANSI/ASME CP-189-1991, "ASNT Standard for Qualification and Certification of Nondestructive Testing Personnel," dated March 15, 1991

Report SG-SGDA-02-026, "Fort Calhoun Station Steam Generator Operational Assessment For Cycle 21, Spring 2002," dated July 2002

Report SG-SGDA-03-16, "Fort Calhoun Station Steam Generator Degradation Assessment," Revision 1 dated September 2003

OPPD Letter LIC-98-0031, Response to Generic Letter 97-05, dated March 16, 1998

Welding Procedure Specification, WPS 801, Gas Tungsten Arc Welding Procedure used for welding ASME Code Section III, Class 2, 2" stainless steel piping and socket welds

**Section 2OS1: Access Control to Radiologically Significant Areas**

Condition Reports

200201480, 200201606, 200201727, 200201967, 200202250, 200300325, 200300365, 20030652, 200300922, 200301418, 200302044, 200303643, 200303664, and 200303672

Procedures

RP-204, “Radiological Area Controls,” Revision 34
RP-502, “Use of Respiratory Protection Equipment,” Revision 13
SO-O-47, “Spent Fuel Pool Inventory Control,” Revision 4
SO-G-101, “Radiation Worker Practices,” Revision 22

RWPs

| | |
|---|---|
| 03-1012 | CVCS Support Modification, Revision 1 |
| 03-1502 | Minor Maintenance Support 2003 RFO, Revision 0 |
| 03-2520 | Small Bore Piping Modification, Revision 0 |
| 03-3532 | Retrieval of Flashlight from the “B” Steam Generator, Revision 0 |
| 03-3534 | Support the Draining of the Reactor Cavity, Revision 0 |

Self-Assessment and Quality Verification

Quality Assurance Audit Report 58 - Radiation Protection

**Section 2OS2: ALARA Planning and Controls**

ALARA Packages

| | |
|---|---|
| 03-12 | Reactor Head Work (RWP 1512, 2512, 3512-AB) |
| 03-31 | RCS Thermal Expansion Loop Modification (RWP-3548-AB) |
| 03-34 | SI-208 Valve Replacement (RWP 3445, 3446-AB) |

Procedures

FCSG-22, “Outage Planning and Execution,” Revision 8
MD-AD-003, “Preparation of Maintenance Work Document,” Revision 5
SO-R-2, “Condition Reporting and Corrective Action,” Revision 24
RP-201, “Radiation Work Permits,” Revision 24
RP-AD-300, “ALARA Program,” Revision 10
RP-301, “ALARA Job Reviews,” Revision 21
RP-303, “ALARA Cost-Benefit Analysis,” Revision 4

Self-Assessment Quality Verification

Radiation Protection Program 2003 Self-Assessment Report SA-26
Quality Department Surveillance Report #58(1)-0603, "Common Cause Evaluation of Rad. Events"

Condition Reports

200203529, 200303158, 200303953, 200303967, 200304005, and 200304274

Miscellaneous

Fort Calhoun Station Dose Reduction Plan 2003 to 2007

## LIST OF ACRONYMS

| | |
|---|---|
| ALARA | as low as is reasonably achievable |
| ASME | American Society of Mechanical Engineers |
| CFR | *Code of Federal Regulations* |
| LER | licensee event report |
| NCV | noncited violations |
| NRC | Nuclear Regulatory Commission |
| RWP | radiation work permit |