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

May 14, 2004

EA-04-078

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 AND NOTICE OF VIOLATION 05000285/2004002

Dear Mr. Ridenoure:

On March 31, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. The enclosed integrated inspection report documents the inspection findings which were discussed on April 5, 2004, with Mr. Ralph Phelps, Division Manager, Nuclear Engineering, and other members of your staff.

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

Based on the results of this inspection, the NRC identified one violation which is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it are described in detail in the subject inspection report. The violation is being cited because your staff failed to restore compliance within a reasonable time after a violation was identified.

Additionally, the NRC identified two findings that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC also determined that there was a violation associated with one of these findings. This violation is being treated as noncited violation (NCV), consistent with Section VI.A of the Enforcement Policy. This NCV is described in the subject inspection report.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Fort Calhoun Station facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosures, 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/

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

Docket: 50-285
License: DPR-40

Enclosures:
1. Notice of Violation
2. NRC Inspection Report 05000285/2004002
   w/attachment: Supplemental Information

cc w/enclosures:
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NOTICE OF VIOLATION

Omaha Public Power District
Fort Calhoun Station

Docket 50-285
License DPR-40
EA-04-078

During an NRC inspection conducted on January 1 through March 31, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR Part 50, Appendix B, Criterion V, states, in part, that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.
>
> Contrary to the above, on January 21, 2004, the licensee failed to assure that procedures included appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to assure that Procedure OP-ST-DG-0001, "Diesel Generator 1 Check," Revision 39, contained appropriate acceptance criteria for frequency when performing a fast start of the diesel generator. The acceptance criteria did not account for a 2 hertz speed droop of the fully loaded diesel generator when selecting the minimum acceptable frequency. The licensee had previously received a noncited violation (NCV 05000285/2003005-02) as a result of a similar condition.
>
> This is a violation of very low safety significance (Green).

Pursuant to the provisions of 10 CFR 2.201, Omaha Public Power District is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-04-078" and should include: (1) the reason for the violation or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 14th day of May 2004

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285
License:	DPR-40
Report:	05000285/2004002
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	Fort Calhoun Station FC-2-4 Adm. P.O. Box 399, Hwy. 75 - North of Fort Calhoun Fort Calhoun, Nebraska
Dates:	January 1 through March 31, 2004
Inspectors:	J. Kramer, Senior Resident Inspector L. Willoughby, Resident Inspector T. McKernon, Senior Operations Engineer N. O'Keefe, Senior Reactor Inspector
Approved By:	Kriss M. Kennedy, Chief, Project Branch C Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2004002; 01/01/2004 - 03/31/2004; Fort Calhoun Station, Integrated Resident and Regional Report; Surveillance Testing, Problem Identification and Resolution, Other

The report covered a 3-month period of inspection by Resident and Regional office inspectors. One Green cited violation, one Green noncited violation, and one Green finding were identified. The significance of most 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. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A violation of 10 CFR Part 50, Appendix B, Criterion V, was identified as a result of the diesel generator test procedure not containing appropriate quantitative or qualitative acceptance criteria to determine operability of diesel generators when conducting the full speed starts of the diesel generators. The licensee's acceptance criteria did not account for a 2 hertz speed droop of the fully loaded diesel generator when selecting the minimum acceptable frequency. The licensee had previously received a noncited violation (NCV 05000285/2003005-02) as a result of a similar condition adverse to quality.

  This finding was considered more than minor because it was associated with the procedure quality attribute of the mitigating systems cornerstone in that the procedure did not contain appropriate quantitative acceptance criteria to ensure the capability of the diesel generator to meet its design basis requirements. The finding was characterized under the Significance Determination Process as having very low safety significance because the as-found diesel generator frequency and voltage were adequate to support the emergency core cooling system loads and no actual loss of safety function occurred. This finding also had crosscutting aspects associated with problem identification and resolution because the licensee failed to correct a previously identified violation (Section 4OA2.1).

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion III, was identified as a result of not properly translating design requirements into procedures. Procedure AOP-17, "Loss of Instrument Air," Revision 5, did not provide adequate steps to respond to a prolonged loss of instrument air. Select valves were equipped with air accumulators or backup nitrogen supplies to maintain the valves operable after a loss of instrument air. The safety injection refueling water tank recirculation valves have a 30-day design mission time during a loss-of-coolant accident, but were provided with an accumulator capable of lasting 39 hours. If the accumulator were to become expended, the valves would fail open and divert water from containment recirculation to the safety injection refueling water tank.

This finding was more than minor because it was related to the equipment performance availability attribute of the mitigating systems cornerstone objective and the design and configuration control attributes of the barrier integrity cornerstone objective. The senior reactor analyst determined that the safety injection refueling water tank recirculation valves would have remained closed throughout the risk-significant portion of their mission time. Additionally, the senior reactor analyst concluded that the likelihood of a loss-of-coolant accident combined with a loss of instrument air was sufficiently small so that further evaluation of the change in risk beyond the modeled mission time was not required. Therefore, the failure to have an adequate abnormal operating procedure for loss of instrument air represented a finding of very low risk significance (Section 4OA5.1).

- Green. A finding was identified as a result of the licensee performing an unauthorized modification to the coupling guard on the diesel-driven auxiliary feedwater pump. Licensee personnel wrapped red duct tape around the guard to reduce the excessive vibration due to broken welds on the guard. Since the diesel-driven auxiliary feedwater pump is not safety related, the unauthorized modification to the coupling guard was not a violation of requirements.

  This finding was more than minor since it is associated with the equipment performance reliability attribute of the cornerstone. The finding was characterized as having very low safety significance because the pump remained available to support unit operations. This finding also had crosscutting aspects associated with human performance because personnel performed an unauthorized temporary modification to a coupling guard (Section 1R22).

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

The unit began the inspection period at 100 percent power. On March 25, 2004, operators commenced lowering power in preparation for a scheduled 10-day midcycle outage. The unit was removed from service the following day. On March 28, the unit entered the first of two planned midloop conditions for the replacement of reactor coolant pump seal packages. The unit exited the midloop condition the following day. On March 31, the unit returned to a midloop condition to complete the repairs to the reactor coolant pump seals. The unit was shut down in Mode 4 at the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

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

- Low Pressure Safety Injection Pump SI-1A while Low Pressure Safety Injection Pump SI-1B was inoperable for maintenance on January 15, 2004

- Diesel Generator 2 air start system while Diesel Generator 1 was inoperable for maintenance on January 21, 2004

- Charging Pumps CH-1A and CH-1B during an outage of Charging Pump CH-1C on March 3, 2004

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors performed routine fire inspection tours (seven inspection samples) and reviewed relevant records for plant areas important to reactor safety. The inspectors observed the material condition of plant fire protection equipment, the control of transient combustibles, and the operational status of barriers. The inspectors compared inplant 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 10 - Charging Pump Area (Room 6)
- Fire Area 20.1 - Personnel Air Lock Door Area (Room 58)
- Fire Area 23 - Pipe Penetration Area (Room 59)
- Fire Area 35A - Diesel Generator 1 (Room 63)
- Fire Area 42 - Main Control Cabinets in Control Room (Room 77)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors performed a review of the flood protection measures (one inspection sample). The inspectors reviewed the Probabilistic Risk Assessment Summary Notebook for internal flooding events. The inspectors performed walkdowns of Corridor 4 and Room 22 to verify that equipment was not subject to damage as a result of internal flooding when the floor plug between Corridor 4 and Room 22 was removed. The inspectors reviewed the internal flooding analysis that demonstrated that the safety-related equipment in other rooms was not vulnerable to this internal flooding.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Quarterly Review of Requalification Activities

a. Inspection Scope

The inspectors performed a licensed operator requalification observation (one inspection sample). On February 23, 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 operating procedures and 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.

.2 Biennial Review of Requalification Activities

a. Inspection Scope

The inspectors reviewed the annual operating examination test results for 2003. Since this was the first half of the biennial requalification cycle, the licensee had not yet administered the written examination. These results were assessed to determine if they were consistent with NUREG 1021, “Operator Licensing Examination Standards for Power Reactors,” Revision 8, Supplement 1, guidance and Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process,” requirements. This review included examination of test results for a total of 51 licensed operators, which included shift-standing senior operators, staff senior operators, shift-standing reactor operators, and staff reactor operators.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

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

- Safety Injection Refueling Water Tank Recirculation Valve HCV-386 (Condition Report 200400169)

- Condenser Evacuation In-Line Gas Radiation Monitor RM-057 (Condition Report 200301279)

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- Outage of Diesel-Driven Auxiliary Feedwater Pump FW-54, High Pressure Safety Injection Pump SI-2B, Charging Pump CH-1A, Air Compressor CA-1C, and Condenser Evacuation Pump FW-8A on January 14, 2004

- Outage of Diesel Generator DG-1, Circulating Water Pump CW-1A, Containment Cooling Coil Inlet Isolation Valve HCV-401B, Normal Range Stack Gas Radiation Monitor Remote Ratemeter RM-062, and Accident Range Stack Gas Radiation Monitor Remote Ratemeter RM-063 on January 21, 2004

- Outage of Boric Acid Pump CH-4A, Charging Pump CH-1A, and Circulating Water Pump CW-1A on February 19, 2004

- Outage risk assessment and management for the unit shutdown and maintenance outage that started on March 26, 2004

b. Findings

No findings of significance were identified.

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

1. Intake Blockage

a. Inspection Scope

On January 29, 2004, operators entered Procedure AOP-1, "Acts of Nature," Revision 14, as a result of high differential pressure across the intake trash grids caused by the accumulation of debris and ice. Operators backwashed the trash grids to remove the debris and also checked whether frazil ice conditions existed at the trash grids.

While backwashing the first trash grid, in accordance with Procedure OI-CW-1, “Circulating Water System Normal Operation,” Revision 37, the operators throttled the surface sluice isolation valve closed to allow a higher backwash pressure to develop. Upon completion of the backwash of the first grid, the operators attempted to reopen the surface sluice isolation valve. A pin in the reach rod assembly to the valve broke and the surface sluice isolation valve remained in the throttled position. This caused insufficient surface sluice flow and allowed surface ice to accumulate on the trash grids. The operators opened the valve locally to establish surface sluice flow which kept the surface ice from accumulating on the trash grids. The operators then proceeded to backwash the remaining trash grids without throttling the surface sluice flow. After backwashing the trash grids several times, the debris and surface ice were removed from the grids and Procedure AOP-1 was exited.

The inspectors observed the backwashing of the trash grids and the evaluation of the icing conditions. The inspectors discussed the event with operations crew involved with the event. The inspectors reviewed Condition Report 200400345 that documented the event (one inspection sample).

b. Findings

No findings of significance were identified.

2. Relay Failure and Loss of Letdown

a. Inspection Scope

On March 22, 2004, a relay in the reactor regulating system failed. The failure caused the pressurizer level Channel Y setpoint to fail low, resulting in the isolation of the letdown system. Operators allowed pressurizer level to rise to the upper end of the control band and then stopped all charging flow into the reactor coolant system. Operators diagnosed the problem and transferred the pressurizer level control to the operable channel (Channel X) and re-established charging and letdown. On March 24, the licensee replaced the failed relay and restored the charging and letdown system to the normal alignment.

The inspectors discussed the relay failure, loss of letdown, and subsequent recovery with the operations crew involved with the event. The inspectors performed a control board walkdown following the event to verify that the plant was stable. The inspectors reviewed the control room logs and Condition Report 200401111 that documented the event (one inspection sample).

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- The adequacy of Restraint SISP-2 located below High Pressure Safety Injection to Reactor Coolant Loop 1B Isolation Valve HCV-311 (Condition Report 200401010)

- The operation of Safety Injection Cooler Outlet Control Valves PCV-2909, PCV-2929, PCV-2949, and PCV-2969 during once through cooling (Condition Report 200401064)

- Required flow from fire pumps to Diesel Generator 2 fire protection sprinkler system (Condition Reports 2002025 and 200400452)

- Diesel Generator 1 start circuitry following fuse replacement (Condition Reports 200400677 and 200400690)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors performed a review of the operator workarounds, the control room deficiency, and control room burden lists. The inspectors focused on the cumulative effects (one inspection sample) of the workarounds on the reliability and availability of mitigating systems and the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspectors reviewed Procedure OPD-4-17, “Control Room Deficiencies, Operator Burdens, and Operator Work Arounds,” Revision 11, that described the programs for handling workarounds and deficiencies. In addition, the inspectors reviewed the licensee’s quarterly assessment of operator workarounds dated February 5, 2004, and the planned corrective actions for the deficiencies.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Tests (71111.19)

a. Inspection Scope

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

- Work Order 00159965-01, replace Diesel Generator 1 Turbo Oil Circulating Pump LO-40-1, Work Order 00141420-02, replace Diesel Generator 1 governor to control rod assembly on January 21, 2004
- Work Order 00108854, install new quick disconnect assemblies on Charging Pump CH-1C on March 2, 2004
- Work Order 00152280-01, replace Diesel Generator Pump Motor LO-33-2-M on March 4, 2004
- Work Order 00139297-01, disassemble, clean, visually inspect, and reassemble Component Cooling Water Heat Exchanger AC-1A on March 15, 2004

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

On March 26, 2004, the licensee entered a planned 10-day outage to replace reactor coolant pump seal packages. The inspectors reviewed the licensee's outage shutdown risk assessment to verify that the licensee appropriately considered risk in planning and scheduling the outage activities. The inspectors observed the plant shutdown and cooldown, the draining to midloop conditions, and shutdown maintenance activities. 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. The inspectors also performed containment tours and verified containment cleanliness.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- OP-ST-CEA-0004, "Secondary CEA Position Indication System Test," Revision 15
- OP-ST-FO-3001, "Diesel Generator 1 Fuel Oil System Pump Inservice Test," Revision 18
- OP-ST-DG-0002, "Diesel Generator 2 Check," Revision 39
- RE-ST-RX-0008, "Shutdown Margin Verification During Hot Shutdown, Cold Shutdown or Refueling," Revision 3
- OP-PM-AFW-0004, "Third Auxiliary Feedwater Pump Operability Verification," Revision 25

b. Findings

Introduction. A Green finding was identified as a result of licensee personnel performing an unauthorized temporary modification to a coupling guard on an auxiliary feedwater pump. Licensee personnel wrapped the coupling guard with red duct tape to reduce vibration.

Description. During operation of Diesel-Driven Auxiliary Feedwater Pump FW-54, licensee personnel observed excessive vibration of a coupling guard. The coupling guard was located between the diesel engine and an electric generator mounted to the front of the engine. The excessive vibration was due, in part, to several broken welds on the guard. As a temporary fix to reduce the vibration, licensee personnel wrapped the guard in red duct tape.

On January 11, 2004, the inspectors observed the coupling guard, with several cracked support welds, wrapped with red duct tape. The inspectors informed the system engineer and shift manager about the observation and questioned the reliability of the pump. Specifically, the inspectors were concerned that the remaining coupling guard welds would break during operation and the coupling would then fall onto the rotating shaft and damage surrounding equipment needed to support pump operation. The licensee acknowledged the inspectors concerns, removed the guard, and placed a temporary barrier around the pump for personnel protection until a permanent repair could be implemented.

Analysis. The inspectors evaluated the safety significance of the finding. The finding affected the mitigating systems cornerstone and was considered more than minor because the modification and degradation of the coupling guard affected the reliability of the pump. The finding was evaluated using the significance determination process as having a very low safety significance because the pump remained available to support unit operations.

This finding had crosscutting aspects associated with human performance. The unauthorized temporary modification to a coupling guard by licensee personnel directly contributed to the finding.

Enforcement. The inspectors evaluated the enforcement aspects of the finding. The finding was associated with a high risk-significant component; however, the diesel-driven auxiliary feedwater pump is not safety-related. Therefore, the unauthorized modification to the coupling guard was not a violation of requirements. This finding (FIN 05000285/2004002-01) was entered into the licensee's corrective action program as Condition Report 200400156.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Modification EC 34182 (one inspection sample) that installed a temporary flange in place of the reactor coolant pump mechanical seal while the mechanical seal was removed for refurbishment. In addition, the inspectors reviewed the 10 CFR 50.59 screening associated with the modification. The inspectors attended the plant review committee meeting that approved the temporary modification. The inspectors performed a walkdown of the modification and verified that the modification had no adverse impact on the safety function of the system.

b. Findings

No findings of significance were identified.

Cornerstones: Emergency Preparedness

1EP6 Drill Observation (71114.06)

a. Inspection Scope

On January 20, 2004, the inspectors observed aspects of the emergency preparedness drill from the simulator and the technical support center (one inspection sample). The purpose of the observations was to evaluate operator performance, licensee event classification, notification of state and local authorities, and the adequacy of protective action recommendations. The inspectors reviewed the licensee's postdrill critiques and discussed observations with licensee management.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

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

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

The inspectors reviewed the performance indicator data for the 4 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.

4OA2 Problem Identification and Resolution (71152)

.1 Diesel Generator Surveillance Testing Procedure

a. Inspection Scope

On January 21, 2004, the inspectors observed operators perform a full speed start of Diesel Generator 1 using Procedure OP-ST-DG-0001, "Diesel Generator 1 Check," Revision 39. The inspectors reviewed the procedure, test data, acceptance criteria, and Technical Specifications. The inspectors evaluated the corrective actions associated with Condition Reports 200302602 and 200302623. The inspectors discussed the diesel generator surveillance testing procedure and the condition report corrective actions with the licensee.

b. Findings

Introduction. A Green violation was identified as a result of the diesel generator test procedure not containing appropriate quantitative or qualitative acceptance criteria to determine operability of the diesel generator as required by 10 CFR Part 50, Appendix B, Criterion V.

Description. NRC Inspection Report 05000285/2003005 documented a Green noncited violation as a result of the diesel generator test procedure not containing appropriate quantitative or qualitative acceptance criteria to determine operability of the diesel generators as required by 10 CFR Part 50, Appendix B, Criterion V. Specifically, for the conduct of the full speed start of Diesel Generator 2 performed on July 7, 2003, to demonstrate that the diesel generator could start and accelerate to rated speed and voltage in less than or equal to 10 seconds without prior warm up, the acceptable frequency band listed in Procedure OP-ST-DG-0002, "Diesel Generator 2 Check," Revision 38, was between 57 and 63 hertz. The inspectors noted that the diesel generator had a 2 hertz frequency droop when going from unloaded to fully loaded during accident conditions based on the governor design. If the diesel generator came up to speed at the minimum acceptable frequency of 57 hertz, as stated in the surveillance procedure, and emergency core cooling systems loads were placed on the diesel generator, the frequency seen by the emergency core cooling system motors would be approximately 55 hertz. This frequency would be outside the ANSI/NEMA MG 1-1998 criteria (57 hertz) for motor operation. The inspectors had previously asked the licensee if all the emergency core cooling system loads would function properly when the diesel generator was running loaded at 55 hertz, the minimum acceptable frequency allowed in the surveillance procedure minus the 2 hertz speed droop. The licensee had indicated that there was no calculation or design basis information to support equipment operability at 55 hertz. The inspectors determined that Procedure OP-ST-DG-0002 did not contain appropriate quantitative or qualitative acceptance criteria to determine operability of the diesel generator. The licensee entered this violation in their corrective action program as Condition Reports 200302602 and 200302623.

On January 21, 2004, the licensee performed a full speed start of Diesel Generator 1 using Procedure OP-ST-DG-0001, “Diesel Generator 1 Check,” Revision 39, to demonstrate that the diesel generator could start and accelerate to rated speed and voltage in less than or equal to 10 seconds without prior warm up. The inspectors found that the procedure contained the same acceptance criteria for frequency, 57 to 63 hertz, as was found in Procedure OP-ST-DG-0002 during the July 2003 test of Diesel Generator 2. Procedures OP-ST-DG-0001 and OP-ST-DG-0002 are identical procedures, except the -0001 procedure applies to Diesel Generator 1, and the -0002 procedure applies to Diesel Generator 2. The inspectors found that neither procedure had been corrected by the licensee following the identification of the noncited violation documented in NRC Inspection Report 05000285/2003005 and that the licensee had failed to assure that a condition adverse to quality was corrected.

Analysis. The inspectors evaluated the safety significance of the finding. This finding affected the Mitigating Systems cornerstone and was considered more than minor because the procedure did not contain appropriate quantitative acceptance criteria to ensure the capability of the diesel generator to meet its design basis requirements. The finding was characterized under the Significance Determination Process as having very low safety significance because the as-found diesel generator frequency and voltage were adequate to support the emergency core cooling system loads and no actual loss of safety function occurred.

This finding had crosscutting aspects associated with problem identification and resolution. The failure of the licensee to correct the procedure when previously receiving a noncited violation for a similar condition directly contributed to the finding.

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, states, in part, that procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, on January 21, 2004, the licensee failed to assure that procedures include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to assure that Procedure OP-ST-DG-0001, “Diesel Generator 1 Check,” Revision 39, contained appropriate acceptance criteria for frequency when performing a fast start of the diesel generator. The licensee had previously received a noncited violation (NCV 05000285/2003005-02) as a result of a similar condition adverse to quality identified on July 7, 2003. This violation of 10 CFR Part 50, Appendix B, Criterion V, is being treated as a violation, consistent with the Enforcement Policy (VIO 05000285/2004002-02). This violation is in the licensee’s corrective action program as Condition Report 200400517.

.2 Reference to Crosscutting Findings Documented Elsewhere in the Report

Section 1R22 describes the unauthorized temporary modification licensee personnel performed on Diesel-Driven Auxiliary Feedwater Pump FW-54. Licensee personnel wrapped red duct tape around a vibrating coupling guard with broken support welds.

This finding had crosscutting aspects associated with human performance. The unauthorized temporary modification to a coupling guard by licensee personnel directly contributed to the finding.

4OA3 Event Followup (71153)

(Closed) Licensee Event Report 05000285/2002004-00: Inadequate procedural guidance resulting in noncompliance with 10 CFR Part 50, Appendix R.

During the performance of an NRC triennial fire protection inspection, the team identified a noncited violation 10 CFR Part 50, Appendix R, Section III.G.2, as committed to in License Condition D of the Fort Calhoun Station operating license. Specifically, the noncited violation was for the failure to assure that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control stations is free of fire damage, as required by 10 CFR Part 50, Appendix R, Section III.G.2. The team identified that a fire in either Fire Area 6 or Fire Area 36A could result in the loss of all three trains of charging pumps. The licensee credits these pumps for accomplishing the hot shutdown function of reactor coolant inventory control. The team documented this finding in NRC Inspection Report 05000285/2003002, Section 1R05.2. The licensee event report was reviewed by the inspectors and no new findings were identified. The licensee documented the issue in Condition Report 200204129. This licensee event report is closed.

4OA5 Other

.1 (Closed) Unresolved Item 05000458/2003011-03: Failure to Provide Means to Assure Proper Emergency Core Cooling System Alignment During Prolonged Loss of Instrument Air

Introduction. A Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, was identified. Specifically, the procedure for the loss of instrument air did not contain sufficient information for addressing a loss of instrument air over an extended period.

Description. NRC Inspection Report 05000458/2003011 documented that Procedure AOP-17, “Loss of Instrument Air,” Revision 5, did not contain actions to be taken in the event of a prolonged period without an instrument air source. The plant safety analyses stated that the plant did not need the bulk of the instrument air system to function during design basis events. The safety-related instrument air loads were provided with accumulators or backup nitrogen bottles which were capable of providing pressure for a specified duration in order to maintain the functions required for safe operation.

However, in the case of the Safety Injection Refueling Water Tank Recirculation Valves HCV-385 and HCV-386, accumulators were provided that were designed to hold the valves closed for 13 hours, after which time the valves would fail open. (Test data from 1998 to present indicated the minimum accumulator bleed-down time was

39 hours.) These valves were in series in the recirculation line that was common to all trains of emergency core cooling systems, allowing water to return to the safety injection refueling water tank. At the beginning of a loss-of-coolant accident, these valves would be in the open position. The valves close when the injection phase is complete and the emergency core cooling systems switch to containment sump recirculation. If these valves lost operating air pressure and failed open during sump recirculation, the borated water used to keep the reactor shut down and to cool the core would be diverted to the safety injection refueling water tank. This would ultimately cause a loss of the ability to run the containment spray pumps and then the safety injection pumps as water level lowered in the sumps.

These valves also served as a containment boundary during the recirculation phase. The increased leakage from the containment to the safety injection refueling water tank would be greater than the limit used in the analysis of Updated Safety Analysis Report, Section 14.15.8, "Radiological Consequences of a LOCA;" therefore, the resulting dose assessment would also be increased.

Analysis. In accordance with Inspection Manual Chapter 0612, this finding was more than minor because it was related to the equipment performance availability attribute of the mitigating systems cornerstone. It affected the cornerstone objective in that the performance deficiency affected the reliability of the emergency core cooling system to respond continuously to the design basis 30-day accident. This finding also affected the design and configuration control attributes of the barrier integrity cornerstone in that the performance deficiency affected the assurance of the containment barrier to be able to protect the public from radionuclide releases caused by accidents.

In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Attachment 1, "User Guidance for Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors determined that a Phase 2 significance determination was required because two cornerstones were affected. However, because two coincident initiating events were required for this finding to affect the plant mitigating capability, the inspectors determined that a Phase 3 evaluation was required.

The senior reactor analyst determined that, while this finding clearly involved an increase in the core damage frequency, the current probabilistic models were not sufficient to quantify this change. However, the likelihood of having a loss-of-coolant accident with a loss of instrument air, combined with emergency response personnel failing to restore instrument air and/or bring the plant to cold shutdown conditions in 39 hours, was very small.

While the design basis requires the emergency core cooling system to operate for 30 days following a design basis accident, the mission time for the risk-significant function of the system is 24 hours. This assumption exists in both the licensee's probabilistic risk assessment and the NRC's Standardized Plant Analysis Risk Model for Fort Calhoun Station. This assumption was developed considering that: decay heat is lower 24 hours into an accident, providing additional time to respond to degrading conditions; the demands on the systems are less; and the Technical Support Center is fully manned and capable of assisting plant staff in making affective repairs to plant equipment.

The senior reactor analyst determined that the safety injection refueling water tank recirculation valves would have remained closed throughout their risk-significant mission time. Additionally, the senior reactor analyst concluded that the likelihood of a loss-of-coolant accident combined with a loss of instrument air was sufficiently small that further evaluation of the change in risk beyond the modeled mission time was not required. Therefore, the failure to have an adequate abnormal operating procedure for loss of instrument air represented a finding of very low risk significance (Green).

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the measures established to assure that design information was correctly translated into procedures were inadequate. Specifically, the requirement to maintain air-operated valves in their required positions during a loss-of-coolant accident with instrument air unavailable was not addressed beyond the initial system responses when backup air accumulators and nitrogen supplies are available. This violation of 10 CFR Part 50, Appendix B, Criterion III, is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000285/2004002-03). This violation was entered into the licensee's corrective action program as Condition Report 200305311.

.2 Temporary Instruction 2515/153: Reactor Containment Sump Blockage (NRC Bulletin 2003-01)

The objective of the Temporary Instruction was to support NRC review of the licensee's activities in response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors (PWRs)."

a. Inspection Scope

The inspectors reviewed the licensee's response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," documented in Letter LIC-03-0105 dated August 3, 2003. In addition, the inspectors reviewed NEI (Nuclear Energy Institute) 02-01, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments," Revision 1, and Condition Report 200302218 that documented the licensee's response to the bulletin. The inspectors discussed the commitments of Letter LIC-03-0105 with licensee personnel.

b. Findings and Observations

The licensee elected to implement method two of NRC Bulletin 2003-01. By implementing method two, the licensee developed interim compensatory measures to reduce the risk which may be associated with potentially degraded or nonconforming emergency core cooling system recirculation functions while performing evaluations to determine compliance. In addition, the licensee has initiated preparations to modify the sumps in case the evaluation concludes that modifications are required. The inspectors reviewed the licensee's response to NRC Bulletin 2003-01 and noted that the licensee had addressed the recommended six interim compensatory measures.

The inspectors reviewed portions of the completed commitments. The inspectors observed the licensed operator training on the identification of the symptoms of a degraded sump during a loss-of-coolant accident. The inspectors reviewed Procedure "EOP/AOP Attachments," Revision 16, and noted that the licensee had developed Attachment 25, "Methods For Refilling the SIRWT Post RAS." During the refueling outage in the fall of 2003, the inspectors observed that the licensee was more aggressive than previous outages in controlling foreign material and debris inside containment.

On September 25 and 27, 2003, the licensee performed a walkdown of containment to verify that drainage paths for recirculation were unblocked. The licensee documented the results in Condition Report 200302218 and concluded that the drainage paths were acceptable and unblocked.

The licensee inspected the containment emergency sumps to verify that they were free of adverse gaps and breaches. The licensee identified some minor gaps in the screens to both of the sumps and initiated repairs to the screens. The licensee documented the observation in Condition Report 2000304391 and concluded that the sumps were operable in the as-found condition. The inspectors reviewed Condition Report 2000304391 and concluded that the minor gaps did not affect the system operability and documented the results in NRC Inspection Report 05000285/2003006, Section 1R15. The inspectors performed a walkdown of the sumps and did not identify any additional adverse gaps or breaches.

4OA6 Meetings

Exit Meeting Summary

On March 31, 2004, the inspector presented the results of the significance determination assessment discussed in Section 4OA5.1 to Mr. G. Cavanaugh, Supervisor, Station Licensing, by telephone, who acknowledged the findings.

The results of the resident inspector activities were presented to Mr. R. Phelps, Division Manager, Nuclear Engineering, and other members of licensee management on April 5, 2004. The licensee's management acknowledged the inspection findings and stated 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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bannister, Plant Manager
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, Division Manager, Nuclear Operations
H. Sefick, Manager, Security and Emergency Planning

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000285/2004002-02	VIO	Inadequate Diesel Generator Surveillance Test Procedure Acceptance Criteria (Section 4OA2.1)

Opened and Closed

05000285/2004002-01	FIN	Unauthorized Modification to the Diesel-Driven Auxiliary Feedwater Pump (Section 1R22)
05000285/2004002-03	NCV	Inadequate Procedure for Long-term Loss of Instrument Air (Section 4OA5.1)

Closed

05000285/2002004-00	LER	Inadequate procedural guidance resulting in noncompliance with 10 CFR Part 50, Appendix R (Section 4AO3)
05000285/2003011-03	URI	Failure to Provide Means to Assure Proper Emergency Core Cooling System Alignment During Prolonged Loss of Instrument Air (Section 4OA5.1)