APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-285/92-29

Operating License: DPR-40

Licensee: Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247

Facility Name: Fort Calhoun Station

Inspection At: Blair, Nebraska

Inspection Conducted: October 11 through November 21, 1992

Inspectors: R. Mullikin, Senior Resident Inspector
R. Azua, Resident Inspector

Approved: P. H. Harrell, Chief, Technical Support Staff, Division of Reactor Projects — Date: 12/4/92

Inspection Summary

Areas Inspected: Routine, unannounced inspection of onsite followup of events, operational safety verification, maintenance and surveillance observations, followup on corrective actions for a violation, and onsite followup of licensee event reports.

Results:

- The licensee's actions in response to the loss of the safety Channel A nuclear detector were proper and exhibited a high degree of concern for safety (Section 2).
- Operations, radiological protection, and security personnel were observed to be performing their duties in an excellent manner (Section 3).
- The housekeeping in the vital areas was found to be very good. However, in the turbine building, several areas require additional attention (Section 3.2).
- Maintenance work activity to replace flange seals on a nonsafety-related valve exceeded the skill of the craft when the activity began to affect safety-related equipment in the vicinity. The work instruction provided

was general in nature and did not provide caution statements regarding the safety-related equipment located in the area (Section 4).

- Prestaging of equipment by maintenance personnel prior to beginning work in a contaminated area demonstrated good radiological protection practices (Section 4).
- Surveillance test procedure adequacy and procedural compliance were found to be very good (Section 5).

Summary of Inspection Findings:

- Violation 285/91[illegible]-01 was closed (Section 6).
- Licensee Event Reports 90-015, 91-015, 91-018, 91-027, 91-028, 91-029, 91-030, 92-009, 92-011, 92-015, 92-019, 92-021, 92-023, and 92-028 were closed (Section 7).

Attachment:

- Attachment - Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

The licensee operated the Fort Calhoun Station at 100 percent power throughout this inspection period.

2 ONSITE RESPONSE TO EVENTS (93702)

2.1 Channel A Excore Detector Failure

On October 25, 1992, the Channel A excore detector, which provided input to the reactor protection system, failed. The licensee initially suspected that a power supply had failed, but troubleshooting determined that the detector had failed inside containment. This detector supplied inputs to the trip units for high power level, thermal margin/low pressure, and axial power distribution. When the detector was determined as the cause, the licensee appropriately entered a 7-day shutdown action statement, as required by the Technical Specifications.

The replacement of the detector would have required a plant shutdown; however, the licensee decided to swap the safety channels with the nonsafety control channel detectors. This would not require a plant shutdown nor a containment entry. The licensee developed the following action plan to proceed with the swapping of nuclear detectors:

- Prepared Engineering Analysis EA-FC-92-78 and a 10 CFR Part 50.59 analysis to address swapping safety Channels A and D with nonsafety control Channels A and B, respectively.

- Issued Operations Memorandum 92-10 to restrict control rod movement until testing was completed on the swapped safety channels.

- Completed the swapping of the channels per an approved temporary modification.

The licensee successfully completed the swapping of the safety channels with the nonsafety control channels on October 30.

2.2 Conclusions

The licensee's actions in response to the loss of safety Channel A nuclear detector were proper and exhibited a high degree of concern for safety.

3 OPERATIONAL SAFETY VERIFICATION (71707)

3.1 Routine Control Room Observations

The inspectors observed operational activities throughout this inspection period to verify that proper control room staffing and control room professionalism were maintained. Shift turnover meetings were conducted in a manner that provided for proper communication of plant status from one shift to the other. Discussions with operators indicated that they were aware of plant and equipment status and reasons for lit annunciators. The inspectors observed that Technical Specification limiting conditions for operation were properly documented and tracked. Operators were observed to properly control access into the control room operating area. Plant management was observed in the control room on a daily basis.

The inspector reviewed the control room log books for danger tags, caution tags, and locked valve deviations. All the logs were observed to be complete. On November 11, 1992, the inspector selected examples from each log to verify that the component reflected the condition stated in the log book. The inspector verified that the correct tags were properly hung on the component for Tagouts 90-0068, 92-0253, 92-0255, 92-2187, 92-2235, and 92-2314. The inspector also verified that two valves (MS-103 and SI-342) were returned to their locked-closed position, as indicated in the locked-component deviation log.

3.2 Plant Tours

The inspectors toured various areas of the plant to verify that proper housekeeping was being maintained. The housekeeping in the vital areas was found to be very good. However, in the turbine building, many areas required attention. The licensee had also identified the housekeeping concerns and has formulated an action plan to solve the housekeeping problem in the turbine building.

The inspectors verified, during plant tours, that various valve and switch positions were correct for the current plant conditions. Personnel were observed obeying rules for personnel safety and rules for escorts, visitors, entry, and exits into and out of vital areas.

3.3 Radiological Protection Program Observations

The inspectors verified that selected activities of the licensee's radiological protection program were properly implemented. Radiation and contaminated areas were properly posted and controlled. Health physics personnel were observed routinely touring the controlled areas. The inspector observed on three occasions that licensee personnel performed the correct process when individuals alarmed the personnel contamination monitor while attempting to exit the radiologically controlled area.

3.4 Security Program Observations

The inspectors observed various aspects of the licensee's security program. Personnel and packages entering the protected area were observed to be properly searched. Nondesignated vehicles entering the protected area were found to be properly escorted by armed security personnel, and security officers were observed performing their tours and/or manning their assigned posts. Compensatory measures were observed to be properly performed whenever a security barrier was inoperable.

On November 17, 1992, the plant security system was unavailable due to hardware changes to the security computer system. This required security officers to be posted at vital doors to unlock the doors and manually log all personnel entering/exiting vital areas. The inspector noted that the officers maintained proper control over the vital area doors and continued to observe the doors for a period after the security computer was returned to service.

3.5 Conclusions

Operations, radiological protection, and security personnel were observed to be performing their duties in an excellent manner.

4 MAINTENANCE OBSERVATIONS (62703)

4.1 Steam Generator Blowdown Control Valve Seal Replacement

On November 10, 1992, the inspector witnessed the maintenance activity that was performed to replace the flange seals on Steam Generator B Blowdown Control Valve HCV-1389. This work activity was controlled by Maintenance Work Order 924494 and its associated work instruction. The maintenance work order had been reviewed and approved, as noted by the appropriate signatures. The inspector reviewed the maintenance work order and determined that the information provided was accurate in identifying the item to be worked on, with specific postmaintenance testing requirements. The work instruction was found to be general in nature, relying more on the skill of the craft.

The work was performed in the lower mechanical penetration room (Room 13), located in the auxiliary building. Due to the potential for contamination when the valve was removed, the area was required to be roped off by radiological protection personnel. Prior to this, the licensee prestaged the equipment needed and removed all applicable piping insulation. The maintenance personnel performing the maintenance activity signed in under the appropriate radiation work permit and wore the required protective clothing as delineated in the radiation work permit.

Prior to initiating the work, the maintenance personnel noted that a hoist would be required to support the valve once it had been disengaged from the pipe. As a result, the maintenance employee, working within the roped-off area, began tying slings to the steam generator blowdown line snubber support and the high pressure safety injection pumps' (SI-2A and -2C) alternate suction line Snubber SI-104. The inspector questioned whether an engineering evaluation had been performed to determine if this was an acceptable practice since some of the equipment involved was safety related. The licensee employee stated that none had been performed but that he did not think that it would be a problem. At this point, the licensee employee decided to halt the work in progress and contacted the system engineer for guidance. Following the discussion with the system engineer, a note was added to the work instruction stating that the slings used to support the hoist could be hung off of the steam generator blowdown line snubber support and a component cooling water pipe located in the vicinity and that this would have no impact on the performance of these systems. The inspector witnessed the remainder of the maintenance activity and noted that the maintenance personnel performed this effort in an appropriate manner. Good radiological protection practices were noted in the removal of equipment, from the contaminated area, by the maintenance personnel. Good health physics coverage was also noted throughout this activity.

4.2 <u>Conclusions</u>

Overall, performance of maintenance personnel was found to be good, with good adherence to radiological protection practices with regard to the removal of equipment from the contaminated area. It must be noted, though, that the determination as to when and if it is appropriate to hang equipment from piping or piping supports was beyond the skill of the craft during this activity. Although the valve that was worked on was in a nonsafety system, consideration upon writing the work instructions for the maintenance work order should have taken into consideration the fact that safety-related equipment in the vicinity could be adversely affected by the maintenance activity.

5 SURVEILLANCE OBSERVATIONS (61726)

5.1 <u>Reactor Anomalies</u>

On November 5, 1992, the inspector witnessed the performance of Surveillance Test Procedure OP-ST-RX-0001, "Reactor Anomalics." This surveillance is performed weekly to satisfy the requirements of Technical Specification 3.10(1)b for the comparison of the overall core reactivity balance to predicted values.

This test can be performed by either a licensed operator or the shift technical advisor. The surveillance witnessed was performed by the shift technical advisor. The inspector verified that the latest revision was being used and that the procedure was being followed. The inspector performed the test independently and the results matched those obtained by the licensee. All results were within the procedural acceptance criteria.

5.2 Conclusions

Surveillance test procedure adequacy and compliance were found to be very good.

6 FOLLOWUP ON CORRECTIVE ACTIONS FOR A VIOLATION (92702)

6.1 (Closed) Violation 285/9121-01: Failure to Take Adequate Corrective Action

This violation concerned the licensee's failure to promptly correct an identified condition in that a station battery jar was discovered cracked on July 1, 1991, but no corrective action was taken until a similar event occurred on September 11. The licensee declared both station batteries inoperable on September 12 and instituted a plant shutdown.

A root cause analysis was completed on August 12, after the battery jar crack discovery on July 1. This analysis concluded that the cracking was caused by stresses in the jar cover due to corrosion buildup around the positive post. The root cause analysis recommended that the batteries be replaced with those having an improved terminal post seal design. Thus, information was available to management on August 12 that a potential common mode failure existed. However, the Plant Review Committee did not meet to discuss the battery cracking problem until August 29. At that meeting, the Plant Review Committee Chairman directed that an operability determination be presented at the next scheduled meeting. This operability determination was not performed.

The licensee attributed the failure to take prompt corrective action on the inadequate application of programs and procedures for identification and correction of adverse conditions. The level of significance after the July 1 cracking event was influenced by past experience, vendor guidance, and engineering judgment. The licensee had performed an engineering evaluation after a similar crack was discovered in March 1991. This evaluation concluded that there was not an operability concern. Thus, the July 1 cracking was influenced by this previous evaluation, which had significant vendor input. In addition, the Plant Review Committee Chairman's directive to perform an operability determination was not performed since it was not tracked after the meeting.

The licensee's corrective actions included revising Nuclear Operations Division Quality Procedure NOD-QP-19, "Root Cause Analysis Guideline." The revisions included the following:

- Vendor information, if critical to the analysis, is required to have the vendor analysis documented.
- Potential common-mode failures should be discussed in the root cause analysis.

- Review of equipment history records for similar failures is required.
- The approved root cause analysis must be forwarded to the Plant Review Committee within 7 days.

In addition, Nuclear Safety Review Group Procedure NSRG-3, "Reviews and Investigations," was revised to enhance their review of root cause analysis reports.

The licensee's actions should provide a quicker and more thorough review of deficient conditions. In addition, items requiring followup from a Plant Review Committee meeting are tracked and assigned the highest action priority. These improvements in the licensee's program should be sufficient to minimize the possibility of further occurrences.

7 ONSITE REVIEW OF LICENSEE EVENT REPORTS (92700)

7.1 (Closed) Licensee Event Report 285/90-015: Nonconservative Setpoints for the Low Temperature/Overpressure Protection System

This report described how the variable setpoints for the two power-operated relief valves used for low temperature/overpressure protection were nonconservative. The licensee determined that the cause of this nonconservatism was deficiencies in the design process for the low temperature/overpressure protection system. The licensee determined that no historical conditions existed that would have had an impact on reactor coolant system integrity due to the nonconservative setpoints.

The licensee's engineering department designed the variable setpoint system for the relief valves in 1984. However, no consideration was given on how the system would operate during a pressure transient such as the inadvertent operation of a reactor coolant pump at low temperatures. The licensee determined that the cause of this event was a lack of adequate design review and an interface between the design group and the technical support department.

The licensee had reorganized the engineering department and improved procedural guidance on the preparation of design packages in 1988. These changes improved the communications problem that contributed to the event. The licensee also established operational limits (pressure-temperature) to ensure reactor coolant system integrity. Other corrective actions included proceduralized requirements for reactor coolant pump restart and high pressure safety injection pump operating criteria.

On June 1, 1992, the licensee submitted an application for amendment of the operating license. This application provided for improved controls on low temperature/overpressure protection. This amendment had not been approved at the end of this inspection period.

7.2 <u>(Closed) Licensee Event Report 285/91-015: Radiation Monitor RM-060 Inoperable Due to Seismic Concerns</u>

This report described the licensee's conclusion, on February 28, 1991, that the flow totalizer for Radiation Monitor RM-060 was nonseismically supported inside of the monitor's cabinet. Monitor RM-060 is the plant stack iodine monitor.

The licensee determined that the cause of the event was the failure to analyze for seismic considerations during the development of the modification for installation of the flow totalizer in 1977. The licensee determined the safety significance of this event to be minimal since Monitor RM-060 is one of five radiation monitors that can initiate a ventilation isolation actuation signal. In addition, the operability of Monitor RM-060 is solely designed for iodine monitoring and accountability. It is also not required by the Technical Specifications for initiation of a ventilation isolation actuation signal.

The licensee's corrective action was to seismically support the flow totalizer on the outside of the cabinet. This was completed on July 13, 1991, and the radiation monitor was declared operable. The inspector reviewed the licensee's completed work and found the flow totalizer to be securely installed on the monitor cabinet. The licensee has made improvements to the modification control process since this flow totalizer was installed in 1977. These improvements should minimize the possibility of a similar occurrence of this type.

7.3 <u>(Closed) Licensee Event Report 285/91-018: Inoperable Station Batteries Due to Inadequate Design of Terminal Post Seals</u>

This report documented the discovery, on September 11, 1991, of a crack in the front wall of a station battery cell. This crack, in addition to previous cracks discovered, prompted the licensee to declare both station batteries inoperable and shut down the plant.

The licensee's root cause analysis found an inadequate design of the battery cell terminal post seals. This design did not adequately allow for the buildup of corrosion products at the positive terminal. The growth of the corrosion radially outward created stresses on the plastic cell jar, which resulted in the cracking.

The licensee shut down the plant and temporarily replaced all of the battery cells with a newer post seal design. The new batteries were obtained from various sources, which included other nuclear facilities, until completely new batteries could be ordered and installed. This was completed and the plant was restarted. Procurement of completely new batteries was begun and the new batteries were installed during the 1992 refueling outage. The inspectors reviewed the battery replacements during both occurrences and noted no problems.

7.4 (Closed) Licensee Event Report 285/91-027: Violation of Containment Integrity by Opening Valve WD-1060 During Sampling

This report documented the licensee's discovery, on November 18, 1991, that containment integrity had been violated when samples were taken from the reactor coolant drain tank discharge line. The sampling was done through Valve WD-1060, which is a drain valve between the two containment isolation valves. Valve WD-1060 was used as a sampling point on 20 different occasions during the investigation of abnormal increases in reactor coolant drain tank level.

NRC Violation 285/9126-02 was cited as a result of this event. The corrective actions were reviewed during the closeout of this violation in NRC Inspection Report 50-285/92-14. This licensee event report is closed.

7.5 (Closed) Licensee Event Report 285/91-028: Unmonitored Release on Loss of the 161-kV System

This event concerned the licensee's discovery, on December 1, 1991, that the sample pump for the exhaust stack gas, iodine, and particulate monitors in the laboratory and radioactive waste processing building was not running. This constituted an unmonitored release since release from the building had been made while the pump was not in operation. The cause was determined to be due to the previous day's loss of the 161-kV offsite power supply resulting from severe winter weather. The design of the power supply to the sample pump and the exhaust fans, where the sample is taken from, is that when power is restored the fans restart, but the sample pump had to be started locally. The licensee determined that the routine releases that occurred while the sample pump was inoperable had minimal safety significance.

The licensee's initial corrective action was to alert operators to the need to reset the power supply to the sample pump upon a loss of power. In addition, a temporary modification (TM-04) was installed, which would automatically restart the sample pump when power is restored. The inspector reviewed this temporary modification and found it would accomplish this purpose.

The licensee's long-term corrective action was to implement Engineering Change Notice 92-524, which made the temporary modification permanent. In addition, it provided control room annunciation upon the loss of power to the sample pump. The inspector reviewed the licensee's corrective actions and found them to be sufficient to address this licensee event report.

7.6 (Closed) Licensee Event Report 285/91-029: Personnel Air Lock Leak Rate Test Deficiency

This licensee event report described a condition that was discovered, on December 2, 1991, by a special services engineer The engineer, while reviewing procedures for containment leak rate testing, identified that the

Type B test procedure for the personnel air lock did not adequately test the inner personnel air lock equalizing valve, as required by Technical Specification 3.5.(3)d.

The root cause of this event was attributed to an inadequate procedure change review process that was used when a change was made to the testing procedure in 1974. Contributing causes to this event included: (1) the lack of understanding or knowledge of the regulations regarding the design basis involved, by all individuals who reviewed and/or approved the procedure change, as well as, by those individuals who have performed the biennial reviews for this procedure; and (2) the fact that in 1974 no procedural requirement existed for performing documented safety evaluations for procedure changes.

The following are corrective actions that were implemented by the licensee immediately following the discovery of the test deficiency and the corrective actions that were implemented to preclude recurrence of this event:

- On December 4, 1991, the inner personnel air lock door equalizing valve was declared inoperable due to the lack of proper leak rate testing. Administrative controls were established to ensure that containment integrity was maintained by danger tagging the outer door closed.

- A procedure (IC ST-AE-0006) was developed to leak test the inner door equalizing valve. This test was approved and incorporated into station procedures on December 6. On December 7 this test was performed with acceptable test results.

- The safety evaluation and review process for procedure changes has been substantially upgraded since 1974 and is documented in Nuclear Operations Division Quality Procedure NOD-QP-3, "10 CFR 50.59 Safety Evaluations."

- The biennial review process has been upgraded as part of an overall enhancement and is documented in Standing Order SO-G-36, "Biennial Review."

- By June 15, 1992, all Type B leak rate test procedures were reviewed, along with the current configuration of all Type B penetrations, to ensure that the penetrations were being tested in accordance with 10 CFR Part 50, Appendix J, criteria.

The inspector reviewed the documentation for the completion of the corrective actions. As a result of the completed actions, this licensee event report is closed.

7.7 <u>(Closed) Licensee Event Report 285/91-030: Radiation Monitors Out of Service With Containment Pressure Reduction in Progress</u>

This report documented the December 10, 1991, event when a containment pressure reduction was in progress while the auxiliary building ventilation stack iodine, particulate, and gas radiation monitors (RM-060, -061, and -062, respectively) were removed for filter replacements. The shift supervisor directed, during the shift briefing, a licensed operator to terminate the containment pressure reduction early before filter replacements began. However, the licensed operator became busy on another task and forgot to terminate the pressure reduction. Another licensed operator, who was unaware of the directive to terminate the pressure reduction, removed the radiation monitors from service for filter replacements. It wasn't until 6 minutes later that an operator realized the Technical Specification requirement was not being met and terminated the containment pressure reduction.

The licensee's evaluation determined that this event resulted from a breakdown in verbal and written communications and work practices. The examples listed by the licensee included:

- The operator that was directed to terminate the containment pressure reduction early did not relay the fact that he was unable to do so.
- The surveillance test procedure used for filter replacements did not require the shift supervisor's approval prior to removing the radiation monitors from service
- Operating Instruction OI-RM-1, "Radiation Monitoring Normal and Accident Operation," did not ensure that releases had been secured prior to removing the radiation monitors from service.
- The operators involved failed to verify that the containment pressure reduction had been terminated prior to filter replacements.

The corrective actions taken by the licensee included the discussion of this event in licensed operator requalification training and procedural revisions. Surveillance Procedure CH-ST-VA-0001, "Auxiliary Building Exhaust Stack Alpha, Iodine and Particulate Sampling and Analysis," was revised to require a shift supervisor's signature prior to removing the radiation monitors from service. In addition, Operating Instruction OI-RM-1 was revised to require the securing of releases prior to removing gas and particulate monitors from service for filter replacement. The inspector reviewed the licensee's corrective actions and found them to be sufficient to close this licensee event report.

7.8 <u>(Closed) Licensee Event Report 92-009: Unplanned Actuation of the Ventilation Isolation Actuation Signal</u>

This event occurred, while in a refueling outage, when electricians replaced fuses for Channel B of the ventilation isolation actuation signal after completing maintenance. The maintenance that being performed was the cleaning and repair/replacement of the lockout relays for the five radiation monitors for Channel B of containment radiation high signal. If any one of the five radiation monitors reaches its setpoint, a containment radiation high signal is generated, which produces a ventilation isolation actuation signal. In this event, one of the five lockout relays was in the tripped condition when fuses were reinstalled. All equipment operated as designed when the inadvertent actuation signal was received.

The licensee determined the root cause of this event to be an inadequate verification and validation of Procedure EM-RR-EX-0201, "Repair/Replacement of Lockout Relays." This procedure did not require that the lockout relays be in a particular position (reset) prior to reinstalling fuses. Also, the maintenance work order instructions did not specify this requirement. An over reliance on the technical adequacy of the procedure and maintenance work order was determined to be a contributing cause.

The licensee's corrective actions included personnel training on this event and the revision of Procedure EM-RR-EX-0201. The inspector verified that the procedure was revised to include the proper lockout relay condition prior to energizing the relays.

7.9 (Closed) Licensee Event Report 92-011: Unacceptable Valve Arrangement for Service Air System Containment Penetration M-74

This event resulted from the discovery, during the 1992 refueling outage, that the compressed air containment penetration valve arrangement did not meet the isolation criteria required for a containment atmosphere-exposed system. Containment Penetration M-74 consisted of an outboard automatic isolation valve (HCV-1749) and an inboard normally-open manual valve (CA-555). For this arrangement to be acceptable, the compressed air system pressure would have to be greater than the maximum containment design pressure. However, since the air compressors would not automatically load onto the emergency diesel generators under accident conditions, the compressed air system pressure would be less than the maximum containment pressure. Thus, the valve arrangement was not acceptable.

The licensee determined that the original Final Safety Analysis Report stated that one air compressor would automatically start during accident conditions. A revision to this report, in 1971, stated that a compressor would not automatically start. However, the containment penetration for the compressed air system was not modified.

The licensee's initial corrective action was to remove Valve CA-555 and install a blank flange before starting up from the refueling outage. This alleviated any immediate safety concern. In addition, the licensee reviewed other valve arrangements for containment penetrations and found no other similar problems. The licensee plans to install a qualified inboard isolation valve during the next refueling outage. This licensee event report is closed based on the licensee's completed and proposed corrective actions.

7.10 (Closed) Licensee Event Report 92-015: Loss of Shutdown Cooling Flow Control and Flow Indication

This report documented the April 12, 1992, event of power being lost to the shutdown cooling flow control valve controller and shutdown cooling flow indication. The plant was in a refueling outage at the time of the event and in an abnormal electrical alignment. The control room operators determined the cause of the loss of power and restored shutdown cooling within 7 minutes. During the event, the reactor coolant system temperature increased 6°F.

The NRC issued Violation 285/9209-01 based on this event for an inadequate procedure that allowed the plant to be in the abnormal electrical lineup. This licensee event report is closed based upon the review to be performed of the licensee's corrective actions for the violation.

7.11 (Closed) Licensee Event Report 92-019: Control Element Assembly Drop and Plant Shutdown Due to Clutch Coil Failure

This report documented the May 31, 1992, event when a control element assembly dropped into the core while at 100 percent power. The licensee reduced power to 70 percent to attempt to recover the rod but was unsuccessful. A plant shutdown was initiated per Technical Specification requirements and a Notification of Unusual Event was declared.

The control element drive mechanisms at the Fort Calhoun Station use a rack and pinion mechanism to perform vertical movement of the assembly. The control rod is held in place by an electromagnetic clutch. The licensee investigated the cause of the control rod drop and found that the clutch coil had failed. The coil was replaced and resistance readings on the clutch coils for the other control rods were taken. No problems were noted and the plant was restarted.

The licensee determined the root cause of this event to be the material failure of the clutch coil in Control Element Drive Mechanism 35. A laboratory analysis of the coil concluded that the failure was caused by an electrical short, possibly due to a manufacturing defect or an induced overstress such as a power surge.

The licensee's immediate corrective actions to replace the failed coil and test the remaining clutch coils, prior to plant startup, was conservative. In addition, the licensee has proposed to evaluate potential means of performing predictive maintenance on the clutch coils. The licensee will perform a further examination of the Control Element Assembly 35 clutch coil circuit during the 1993 refueling outage to look for signs of overstress in a series resistor. The completed and proposed corrective actions are sufficient to close this licensee event report.

7.12 (Closed) Licensee Event report 92-021: Failure to Initiate a Fire Watch for an Inoperable Fire Door

This report resulted from the discovery, on June 11, 1992, that the fire door to the charging pump valve room would close but not latch. The broken latch was discovered by a nonlicensed operator during the performance of monthly Surveillance Test Procedure OP-ST-FP-0001, "Fire Protection System Inspection and Test." The operator who made the discovery noted that the latch was broken but failed to notify the shift supervisor or initiate a maintenance work request. Procedure OP-ST-FP-001 requires that it be performed in accordance with Procedure OI-FP-6, "Fire protection System Inspection and Test." The operator did not consider the fire door inoperable since a note on the fire door checklist in Procedure OI-FP-6 stated that all doors shall be closed unless in use or a fire watch is posted. Since the door would close, it was not declared inoperable. On June 13, the shift supervisor reviewed the completed surveillance test but did not consider a fire watch to be needed based on the same note.

However, on June 17, a general maintenance craftsperson, familiar with fire door requirements, noted the broken door latch and contacted system engineering to generate a fire barrier impairment and initiate a fire watch. The applicable system engineer noted that the Technical Specification requirement for instituting a fire watch was not met. The licensee's immediate corrective was to institute a fire watch and repair the latch.

The licensee determined the root cause to be ambiguous instructions contained in the note at the beginning of the checklist for Procedure OI-FP-6. A contributing cause was determined to be the failure of the operator to report his findings to the shift supervisor as required by procedure. The licensee's corrective actions to prevent recurrence included revising Procedure OI-FP-6 and Standing Order SO-G-58, "Control of Fire Protection System Impairments," to clearly define that an operable fire door must have a functioning latch mechanism. In addition, the shift supervisors discussed with their crews the importance of immediately notifying the shift supervisor of any anomalies or deficiencies when performing a surveillance test. These actions are sufficient to close this licensee event report.

7.13 (Closed) Licensee Event Report 92-023: Reactor Trip Due to Inverter Malfunction and Subsequent Pressurizer Safety Valve Leak

This event occurred on July 3, 1992, when the plant tripped on high pressurizer pressure while at 100 percent power. Maintenance on a nonsafety-related inverter resulted in the momentary loss of power to the turbine electrohydraulic control system and the subsequent closure of the turbine control valves. The high pressurizer pressure resulted in Pressurizer Safety Valve RC-142 lifting and failing to fully reseat. This failure of Valve RC-142 resulted in the loss of approximately 20,000 gallons of reactor coolant to the containment sump.

This event resulted in an Augmented Inspection Team being dispatched to the Fort Calhoun Station on July 4. The cause of this event and the corrective actions taken are documented in NRC Inspection Report 50-285/92-18. In addition, a special inspection was conducted from August 24 through September 3, due to the premature opening of Valve RC-142 on August 22. The results of this inspection and the licensee's corrective actions are documented in NRC Inspection Report 50-285/92-21. This licensee event report is closed based on the inspections listed above.

7.14 (Closed) Licensee Event Report 92-028: Partial Loss of Load Resulting in Pressurizer Safety Valve Lift and Subsequent Reactor Trip

On August 22, 1992, the plant tripped on thermal margin/low pressure while at 100 percent power. A failed power converter in the electrohydraulic control system resulted in the partial closure of the turbine control valves. With a partial loss of load, the reactor coolant system pressure increased, but a pressurizer code safety valve (RC-142) lifted prematurely before the plant tripped on high pressure. The thermal margin/low pressure trip occurred as pressure decreased due to the open safety valve. This event was the subject of a special inspection conducted from August 24 through September 3. The results of that inspection and the licensee's corrective actions are documented in NRC Inspection Report 50-285/92-21. This licensee event report is closed.

ATTACHMENT

1. PERSONS CONTACTED

*R. Andrews, Division Manager, Nuclear Services
J. Bobba, Supervisor, Maintenance
J. Chase, Assistant Manager, Fort Calhoun Station
*M. Frans, Supervisor, Systems Engineering
*W. Gates, Vice President, Nuclear
J Geschwender, Station Licensing Engineer
R. Jaworski, Manager, Station Engineering
*R. Johansen, Supervisor, Maintenance Support
*W. Jones, Senior Vice President
*D. Lippy, Station Licensing Engineer
*W. Orr, Manager, Quality Assurance and Quality Control
*T. Patterson, Manager, Fort Calhoun Station
*R. Phelps, Manager, Design Engineering
A. Richard, Assistant Manager, Fort Calhoun Station
*J. Sefick, Manager, Security Services
*C. Simmons, Station Licensing Engineer
F. Smith, Supervisor, Chemistry
*R. Short, Manager, Nuclear Licensing and Industry Affairs
*J. Tesarek, Supervisor, Simulator Services
J. Tills, Operations Supervisor

*Denotes licensee personnel that attended the exit meeting. In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on November 25, 1992. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.