APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-285/93-03

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: January 31 through March 13, 1993

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

Accompanying Personnel: C. Skinner, Intern

Approved: P. H. Harrell, Chief, Technical Support Staff, Division of Reactor Projects — Date 4/5/93

Inspection Summary

Areas Inspected: Routine, unannounced inspection of onsite followup of events, operational safety verification, maintenance and surveillance observations, onsite followup of licensee event reports, and followup on previous inspection findings.

Results:

- The licensee's inadequate postmaintenance testing efforts, following a temporary modification to the reactor protective system, was an indication of lack of attention to detail and resulted in a violation of Technical Specification 2.15(1) (Section 2.1).

- Operations, radiological protection, and security personnel performed their duties in a professional manner (Section 3).

- Communications between control room personnel and personnel performing a surveillance were found to be excellent (Section 5).

Summary of Inspection Findings:

- Violation 285/9303-01 was identified (Section 2.1).
- Licensee Event Reports 90-014, 91-010, 92-007, 92-031, 92-032, and 93-001 were closed (Section 6).
- Inspection Followup Items 285/9036-01 and 285/9123-01 were closed (Section 7).

Attachment:

- Attachment - Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

At the beginning of this inspection period, the Fort Calhoun Station operated at 100 percent power. On February 24, 1993, the licensee reduced power to 98 percent to ensure that peak linear heat rate was maintained within the limits specified in the Technical Specifications. The plant remained at this power level throughout the remainder of the inspection period.

2 ONSITE RESPONSE TO EVENTS (93702)

2.1 Inoperability of Channel D Reactor Protective System Trip Units

On February 24, 1993, the licensee reduced power to 98 percent. Following the power reduction, small xenon oscillations were observed and tracked on all reactor protective system channels. An abnormality was detected in that Channel D axial shape index was tracking exactly opposite of Channel A. Channels A, B, and C all tracked together, as expected. The licensee made the determination that the identified abnormality was not consistent with core conditions, thus a test was performed on the instrumentation response. The Group 4 rods were inserted into the core to 121 inches to provide approximately a 0.003 axial shape index deflection. Channels A, B, and C responded as expected, but Channel D responded in the opposite direction.

The licensee determined the cause of this abnormality to be a hardware problem, which resulted when Power Range Safety Channels A and D were swapped with Power Range Control Channels A and B following the failure of a power range safety channel detector. This effort was performed on October 30, 1992, under Temporary Modification 92-078. The licensee was not sure where the signal cables for the nuclear instrumentation subchannels for Power Range Control Channel B, which was swapped with Power Range Safety Channel D, were reversed.

The power range safety channels provide reactor power indication and inputs to the reactor protective system. The power range safety channel instrument thimbles are located in the biological shield and are spaced every 90 degrees around the core. The dual-section, uncompensated ion chamber neutron detectors convert the neutron flux into an electrical signal. Identical chambers extend along the upper portion and lower portion of the core. Each dual section detector is called a channel and each of the detector chambers are referred to as subchannels. Signals generated from each subchannel are carried by shielded coaxial connectors to one of four power range safety channel drawer assemblies.

The described condition was found to affect three reactor trip units in the reactor protective system Channel D, which utilize the inputs provided by these subchannels. These trip units are high power level (Unit 1), thermal margin/low pressure (Unit 9), and axial power distribution (Unit 12). On

March 1, 1993, the licensee declared Channel D reactor protective system Trip Units 1, 9, and 12 inoperable and began corrective actions under Maintenance Work Request 930783 to install Temporary Modification 92-078, Revision 1, to swap the signal cables for the nuclear instrumentation subchannels.

The licensee determined that, following the modification effort on October 30, 1992, a calibration of the instrument drawer for Power Range Safety Channels A and D was the only postmaintenance activity that was performed. Due to the fact that this effort was performed with the plant at 100 percent power and at steady-state conditions with no deflection in the axial shape index, no discrepancies were noted and, thus, the signal error was not identified. Technical Specification 2.15(1) states that an inoperable channel may be bypassed for 48 hours and, if not returned to operable status within this time frame, the channel must be placed in the tripped condition. Contrary to these requirements, the licensee failed to return the affected channels to an operable status and did not place them in a tripped condition within the required time limit because an adequate postmaintenance test was not performed. This is a violation of NRC requirements (285/9303-01).

2.2 Conclusions

The licensee's failure to perform an adequate postmaintenance test, for determining operability of the power range safety channel, is of concern due to the importance of the reactor protective system. The apparent presumption that the power range control channel had been installed correctly, and the failure to perform a more challenging postmaintenance test, indicated a lack of attention to detail by licensee personnel.

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.

3.2 Plant Tours

3.2.1 Review of Deficiency Tags

On February 11, 1993, the inspector selected a number of various deficiency tags that were attached to equipment throughout the plant. It appeared, based on the tag description, that some of the deficiencies may have been corrected but the tag not removed. The inspector noted the referenced maintenance work request for six selected deficiency tags. The inspector found that all but one of the maintenance work requests were still awaiting completion. However, one had been completed but the deficiency tag had not been removed. In addition, on February 23 while touring the intake structure, the inspector identified another deficiency tag located on Raw Water Pump AC-10B that should have been removed because the deficient condition had already been corrected. The inspector discussed this concern with the Plant Manager, who stated that the Nuclear Safety Review Group had recently performed a review of control room deficiency tags and found some tags that should have been removed. The Plant Manager stated that the maintenance and operations departments were tasked with locating and removing those deficiency tags that require removal. This task was still in progress at the end of the inspection period.

3.2.2 Cable Spreading Room

On February 18, 1993, the inspector toured the cable spreading room to inspect the overall condition of electrical cables, trays, and conduits, as well as proper electrical separation. The inspector noted that the overall condition of the cable trays and the cables, along with housekeeping within the cable trays, was good.

The inspector used the Fort Calhoun Station Updated Safety Analysis Report, Figure 8.5-1, as the criteria for the inspection of proper electrical separation. Figure 8.5-1, Note 7, stated that, in the cable spreading room, cables with EA and EC prefixes may be routed in the same cable tray or raceway, provided they are separated by a metallic barrier. Similar requirements are specified for cables with EB and ED prefixes. Prefixes are used to define segregation requirements from one cable to another. These signify different trains of equipment or different channels. The color of the cable indicates the prefix. Prefix EA corresponds to red, EB to green, EC to yellow, and ED to blue. The inspector found the following two examples where the electrical separation requirement was not in accordance with the Updated Safety Analysis Report:

- Red cables were draped over the metal divider into the section with yellow cables in Tray C1B, between Sections 8 and 9.

- Cable Tray I2, between Sections 5 and 6, was missing an internal metal divider between red and yellow cables.

In addition, the following observations were noted in the cable spreading room:

- A cable in Tray IB4, between Sections 8 and 9, appeared to have an inadequate cable bend radius.

- Four condulet covers were missing near Trays 6P1 and C2A.

- Conduit penetrating the ceiling near Column D-7A did not appear to be adequately supported.

The licensee inspected the cable spreading room and found no other separation concerns but did identify one other missing condulet cover. The licensee issued Maintenance Work Order 930718 to install the missing condulet covers, replace the missing tray divider in cable Tray I2, and investigate the minimum bend radius concern. The condulet covers and the missing tray divider were installed. The inspector verified that the covers and divider were installed. The licensee determined that the cable with the bend radius concern was an abandoned cable. The licensee marked this cable as abandoned.

Maintenance Work Order 930720 was written to place the cables in Tray C1B into their respective sides of the divider. The licensee was able to do this except for one cable. The licensee justified leaving this red cable routed with yellow cables based upon Engineering Analysis EA-FC-90-076, Revision 2. This analysis stated that the instrument and control cables located in the cable spreading room are adequately protected, as is, with the cable jackets and conductor insulation. This analysis was performed after a walkdown of the cable spreading room was performed. The inspector reviewed the engineering analysis and found that it did justify leaving the red cable in place. The Updated Safety Analysis Report stated that deviations from the separation criteria are acceptable, provided an analysis justifies the deviation.

The licensee, in order to address the concern with the conduit support, initiated Engineering Assistance Request 93-034. The conclusion, based on the evaluation, was that the conduit was adequately supported as installed. This properly addressed the inspector's concern.

3.2.3 Safety Injection System Support

On February 25, 1993, the inspector noted, in the west safety injection room, that two seismic supports were in contact with each other. The two supports (SIS-39-BOTTOM and SIS-39A) were on the safety injection and refueling water tank discharge line to Train B of safety injection. The installation of the two supports was such that it appeared that Support SIS-39-BOTTOM could catch on the other support during a seismic event and possibly prevent the support from performing its design function. The licensee was informed of this concern and design engineering responded that adequate seismic support existed for this line even without the contribution of the two supports in question. However, the licensee decided that, to prevent any possible problem with the two supports, the case of Support SIS-39-BOTTOM should be rotated. The licensee created Maintenance Work Order 930792 to perform this work, which had not been completed at the end of the inspection period.

3.2.4 Hearing Protection

On February 23, while touring the turbine building, the inspectors noted that the ear plug dispensers at some of the entry points to the turbine building were empty. In addition, it was noted that a dispenser had not been installed at the first floor entrance to the turbine building from the service building. The licensee explained that this was an oversight due mainly to the fact that several departments at the Fort Calhoun Station had been reorganized and that the personnel who were responsible for replacing the ear plugs had been reassigned to another department and no provisions were made to transfer this specific responsibility. The earplug dispensers were filled and the licensee plans to check and fill the ear plug dispensers on a daily basis.

In addition, the responsibility for this function has been assigned to the supervisor of Plant Administration, who, if he is unavailable, will assign the responsibility to another person in the department.

3.2.5 Emergency Diesel Generators

During this inspection period, the inspectors toured the diesel generator rooms on several occasions. The inspectors noted that on the angle gear drive, which is used to drive the cooling fans, there is a bulls-eye, oil sight glass with an arrow that indicates where the oil level should be maintained. The inspectors identified, that on Diesel Generator 1, the level was right at the arrow and that on Diesel Generator 2 the level was at 3/4 full (above the arrow). The inspectors questioned the system engineer as to whether this was adequate. The system engineer believed that it was adequate but was not able to state what was an acceptable indication. The system engineer contacted the vendor, who stated that, as long as the oil level is visible in the sight glass, it is acceptable.

It must be noted that the item identified above is a visible indication of the capability of the diesel generator to perform its function and, as such, the significance of its indication should have been known.

3.2.6 Air-Operated Valves

While touring the auxiliary building, the inspectors noted that the pressure gauge for Instrument Air Regulator IA-HCV-1387B-FR, located on Steam Generator RC-2B Blowdown Isolation Valve HCV-1387B, appeared to be broken. The face plate was cracked and the gauge was reading 0 psi. Also, the inspectors noted that the pressure gauges for Instrument Air Regulators IA-HCV-385-R and IA-HCV-386-R were reading 50 psi and 40 psi, respectively. These instrument air regulators are part of minimum recirculation to Safety Injection and Refueling Water Tank Isolation Valves HCV-385 and HCV-386. The inspectors questioned the licensee concerning the broken regulator, the difference in pressure indication for two identical valves that perform the same function, and the significance of the regulator gauge indication.

The licensee stated that pressure regulators are set at approximately 5 psi above the bench setting of the valve. For example, for the Blowdown Isolation Valve HCV-1387B valve operator, the manufacturer stated that it takes 16-28 psi to overcome the spring force of the valve, with a 1000-psi pressure difference across the valve, to fully open the valve. Therefore, the pressure regulator should be set for 28 psi + 5 psi (33 psi). The licensee's experience has been that the gauges that are mounted on the regulators do fail occasionally but that the actual regulator setpoints have not shown any appreciable degradation, other than that which may be caused by ambient conditions and which only results is a 1-2 psi change. When a regulator gauge is found to be broken, the regulator is replaced, and the as-found pressure setpoint for the broken regulator is verified. A maintenance work request was issued for the repair of Instrument Air Valve IA-HCV-1387B-FR. It must be noted that this regulator does not perform a safety function because the valve has a fail-closed position on the loss of instrument air and the valve does perform any other safety functions.

With regard to the difference in pressure indication for Valves HCV-385 and HCV-386, the licensee found that the original pressure setpoint for both valves was 40 psi. At one point, the licensee was concerned with the possibility of backleakage through the valve into the safety injection and refueling water tank. The licensee decided to increase the instrument air regulator setpoints of both valves to 50 psi (the lowest allowable setpoint for these valves to perform their safety function is 32 psi and the highest allowable setpoint is 59 psi). Recently, the valve regulator for Valve HCV-386 failed and had to be replaced. During this replacement effort, the instrument air regulator pressure setpoint was set at 40 psi (the licensee had failed to fully document the change in pressure setpoint from 40 psi to 50 psi), thus the difference in readings on the instrument air regulator pressure gauges identified by the inspectors. The licensee noted that, even though the pressure setpoints were different, both valves were capable of performing their safety function. Presently, both valve pressure regulators are set at 50 psi.

The licensee stated that no formal program existed to inspect the instrument air regulator pressure gauges. The licensee replaces these regulators on a regularly scheduled basis as part of the preventive maintenance program. The licensee's experience has been that the regulators have a low failure rate and that, if a regulator were to fail, the failure would be identified during scheduled routine surveillances of the valves.

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 inspectors randomly verified that doors to high radiation areas were locked.

On March 4, 1993, during the maintenance effort on minimum recirculation to Safety Injection and Refueling Water Tank Isolation Valve HCV-386, radiological protection personnel routinely visited the work site to verify radiation levels and to verify that the personnel working on the valve were adhering to good radiological protection practices. In addition, the radiological protection technician also identified areas in which personnel monitoring the maintenance effort should stand to minimize their radiation exposure.

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.

On the night of February 8, 1993, during a period of freezing rain, and on February 12, during a snowfall, the inspector toured the central alarm station to determine if the security cameras were providing adequate coverage of the protected area. The inspector noted that all the security cameras were working and provided a clear view of their assigned areas.

3.5 Conclusions

Operations personnel performance during this inspection period was found to be good. Management involvement in daily plant activities was apparent. The lack of awareness with regard to a visual indication of the diesel generator's capability to perform its function demonstrated a lack of attention to detail. Radiological protection and security personnel performed their duties in a professional manner.

4 MAINTENANCE OBSERVATION (62703)

On March 4, 1993, the inspectors witnessed portions of the licensee's maintenance efforts to inspect and repair Safety Injection and Refueling Water Tank Recirculation Valve HCV-386. This effort was being performed as a result of the valve's failure to meet the acceptance criteria set forth in Surveillance Test Procedure OP-ST-SI-3001, "Safety Injection System Category A and B Valve Exercise Test," because the valve failed to stroke closed in a smooth and unhampered manner. The licensee issued Maintenance Work Order 930316 to identify the source of the problem. During this effort, the licensee inspected the packing and found it to be set at 10 foot-pounds. The licensee determined that this was loose and, thus, increased the packing torque to 38 foot-pounds. While continuing its inspection efforts, a technician inadvertently broke the petcock to the air operated valve's instrument air regulator, thus necessitating its replacement. The licensee wrote Maintenance Work Order 930820 to replace the solenoid valve and Maintenance Work Order 930825 to replace the instrument air regulator.

The inspector reviewed Maintenance Work Orders 930820 and 930825, including their associated referenced Procedures IC-PM-FX-0600, "Air Regulator Filter Maintenance"; EM-RR-VX-0501, "Replacement of ASCO Solenoid Valve"; EM-RR-EX-0600, "Splicing of Electrical Conductors"; and OP-ST-SI-3001, which was used for the postmaintenance testing efforts. The maintenance work orders and the procedures were found to be technically adequate and did not exceed the skill of the craft. These documents had been reviewed and approved, as noted by the appropriate signatures.

The inspector interviewed the instrument and control technicians and the electrical maintenance personnel involved in performing this effort. These members of the licensee staff were found to be knowledgeable of their responsibilities. The licensee held prebriefings on this effort and prestaged equipment to both prevent personnel errors and reduce personnel exposure in an effort to maintain good as-low-as-reasonably-achievable practices. This was not wholly successful because a question regarding the wire splicing material to be used in the installation of the solenoid valve caused the licensee personnel performing this effort to remain in the radiological controlled area until the question was resolved.

Following the completion of the maintenance activity on Valve HCV-386, the licensee performed the stroke test on the valve. When the valve was stroked open, it appeared that the valve stem traveled in a smooth and unhampered manner, but it was noted by one of the technicians that the stem had not traveled its complete distance. Unaware as to the reason why, the licensee issued Maintenance Work Order 937005 to investigate and repair the cause for which the valve would not fully stroke. During this effort, it was identified that, when the valve stem packing was loosened from the 38 foot-pounds that was applied earlier, the valve stem moved within 1/4 inch of full stroke. The licensee proceeded then to check the manual operator for the valve and found that it was not fully open. The manual operator was opened fully (the handwheel rotation was less than a quarter of a turn) and the valve stem moved to the full stroke position. The licensee stated that the handwheel was not fully open because a technician had inadvertently moved the handwheel while performing maintenance on the valve. The licensee performed the postmaintenance test per Procedure OP-ST-SI-3001. The valve was found to meet the acceptance criteria set forth in the procedure.

4.2 Conclusions

Overall performance by maintenance personnel was found to be good, with good attention to detail and procedural compliance. Personnel were knowledgeable of their responsibilities. Work activities were within the skill-of-the-craft. Incomplete preplanning of the maintenance activity resulted in extended stay times and increased radiation exposures.

5 SURVEILLANCE OBSERVATION (61726)

5.1 Diesel Generator 1

On March 3, 1993, the inspectors witnessed portions of the licensee's surveillance activities performed on Diesel Generator 1. This activity was performed in accordance with Attachment 2 of Surveillance Test Procedure OP-ST-DG-0001, "Diesel Generator 1 Check." Proper approvals and tagouts were obtained prior to test initiation. Limiting conditions for operation were met during this surveillance. The surveillance procedure was reviewed and found to have the proper licensee review and approvals, as identified by the appropriate signatures. The procedure was also in conformance with the Technical Specifications.

Personnel performing the surveillance were questioned and were found to be knowledgeable of their responsibilities. Excellent communication between the control room personnel and personnel stationed locally at the diesel generator was noted. Procedural compliance was very good. The inspectors noted that the diesel generator met the acceptance criteria set forth in the surveillance procedure.

5.2 Conclusions

Surveillance activities were properly performed with very good procedural compliance and excellent communications.

6 ONSITE REVIEW OF LICENSEE EVENT REPORTS (92700)

6.1 (Closed) Licensee Event Report 90-014: Component Cooling Water Containment Isolation Valves Outside Design Basis

As a result of NRC Information Notice 89-055, the licensee determined that the component cooling water piping to the reactor coolant pump seal coolers could be susceptible to a high energy line break. The licensee determined that a break of the reactor coolant system hot leg could damage the component cooling water piping and would leave only the outboard containment isolation valve for the component cooling water system to mitigate potential radiological releases.

The component cooling water lines each have an inboard and an outboard containment isolation valve. However, these valves fail in the open position to prevent a loss of cooling to the reactor coolant pump seals. The air-operated, outboard isolation valves are equipped with a nitrogen backup supply to maintain the valves closed during a loss-of-coolant accident, concurrent with a loss of instrument air. The component cooling water system is considered a closed system, as defined in 10 CFR Part 50, Appendix A, provided that it is protected from a loss-of-coolant accident and its dynamic effects. The licensee assumed a failure of the component cooling water lines during a high energy line break due to jet impingement by coolant from the reactor coolant system.

The licensee issued a safety analysis for operability to justify continued plant operation based on a sufficiently low probability of a containment bypass via the component cooling water lines initiated by a loss-of-coolant accident. As corrective action, the licensee planned to finalize the resolution of Unresolved Safety Issue A-2 by utilizing leak-before-break methodology, as per Generic Letter 84-04. The licensee expects this resolution to eliminate the requirement for the postulation of the high energy line break dynamic effects.

The licensee submitted an application for amendment of operating license on February 12, 1993. The proposed Technical Specification change would credit leak-before-break methodology to resolve Unresolved Safety Issue A-2. This licensee event report is closed based on the review of this proposed license change by the Office of Nuclear Reactor Regulation.

6.2 (Closed) Licensee Event Report 91-010: Auxiliary Steam Piping Outside Design Basis

This licensee event report discusses a condition that was identified by the licensee to be outside the design basis for the plant. The condition was related to a 4-foot section of 2-inch, auxiliary steam piping that ran through the upper electrical penetration room. The auxiliary steam system provides heating for plant spaces. The presence of the piping created a condition that could potentially affect redundant, safety-related electrical equipment.

The condition was identified by the licensee while performing walkdowns of piping systems to compile additional information regarding high energy line breaks, so Appendix M of the Updated Safety Analysis Report could be updated. Upon discovery of the steam line, the licensee took immediate actions to ensure that the vulnerability was eliminated by securing the auxiliary steam supply to the plant and by shutting the fire dampers between the rooms that contained the electrical equipment.

An analysis was performed by the licensee and the results indicated that, if a through wall crack occurred in the piping, the rooms containing the electrical equipment would be exposed to an environment of 100 percent humidity and greater than 120°F. These environmental conditions would exceed the qualification limits for the affected equipment.

Although this condition had the potential for impacting redundant electrical equipment, it appears that the condition had minor safety significance. The basis for this determination includes the following:

- The critical crack would have to occur in the steam piping that passes through the electrical equipment room. Since only about 4 feet of piping is in the room, the probability of a through-wall pipe crack occurring at this specific location is extremely small.

- The licensee has not experienced any through-wall cracks in the entire auxiliary steam line since plant operation (approximately 20 years). The steam piping in other areas of the plant is exposed to much more severe environmental conditions than exist in the electrical equipment room.

To permanently eliminate the condition, the licensee removed the piping in the electrical equipment room and rerouted the piping. The piping was rerouted so that a failure in the piping would not affect any safety-related equipment.

6.3 (Closed) Licensee Event Report 92-007: Inadvertent Isolation of Radiation Monitoring During Containment Purge

This licensee event report documented an unplanned actuation of engineered safety feature components. On February 19, 1992, three ventilation isolation valves closed unexpectedly, resulting in the isolation of containment atmosphere process radiation monitors for particulate and noble gas. This event was found by a control room operator, but not until a containment purge was in progress. Technical Specifications require these monitors to be operable and in service during a containment purge.

The licensee determined that it is probable that the event resulted from inadvertent actuation of two relays and the root cause was that the limited work area available within the control room instrument panels caused the craftsman to actuate the relays when performing work inside the panel.

As corrective action, the licensee provided training on Standing Order M-100, "Conduct of Maintenance," to all maintenance personnel on the need for extra precautions when working within the control panels. In addition, an engineering assistance request was initiated to investigate the most appropriate way to alert the operators to inoperability of Radiation Monitors RM-050/051.

The results of the engineering assistance request provided two issues: investigate the most appropriate way to alert operators to the inoperability of Radiation Monitors RM-050/051 and evaluate the incorporation of Ventilation Isolation Valves PCV-742 E/F/G/H interlocks into the radiation monitors sample pump circuitry. These two issues were addressed and incorporated into Modification Request MR-FC-84-155, which will provide a control room annunciator to alarm in the event of a failure of the sample pump for Radiation Monitors RM-050/051 (flow fault). In addition, vacuum relief valves will be added to the sample line between the sample pump and the flow element to prevent the pump from drawing a vacuum, should the isolation valves close, and ultimately cause an alarm due to no flow through the flow element. The licensee plans to incorporate this modification during the 1993 refueling outage.

The inspector reviewed documentation for the completion of the corrective actions taken by the licensee. Based on the review performed by the inspector, it appeared the licensee had taken and plans to take appropriate actions to preclude repetition of this event.

6.4 (Closed) Licensee Event Report 92-031: Inoperability of Fire Suppression Water System

This licensee event report addressed the circumstances that led to the overpressurization of the steam generator blowdown recovery system and resulted in the inoperability of the fire suppression water system. This event, which occurred on December 3, 1992, was addressed in NRC Inspection Report 50-285/92-33 in which a violation of NRC requirements was identified. Routine followup of this violation will address the corrective actions taken by the licensee; therefore, this licensee event report is closed.

6.5 (Closed) Licensee Event Report 92-032: Failure to Satisfy Fire Watch Requirements for Impaired Halon System

This licensee event report described the events that resulted in the failure, by the licensee, to satisfy fire watch requirements for impaired Halon systems. On December 17, 1992, the south door to the east switchgear room was declared inoperable as a fire barrier because of an inoperable latch (i.e., the door could be pushed open without turning the knob). The door serves a security function, as well as being a fire door and a Halon containment boundary, for the east switchgear room. An hourly fire watch patrol was established due to the inoperability of the door. After repair of the door latch was completed and the door declared operable, the licensee determined that a continuous fire watch, with backup fire suppression equipment, should have been established due to the inoperable door representing a degradation of the east switchgear room Halon containment boundary.

The security shift supervisor and the operations shift supervisor involved in this event were interviewed by the licensee and found to be experienced in fire barrier requirements. The security shift supervisor knew the requirements for a degraded Halon containment boundary but did not make the connection for this door. The licensee determined that the root cause of this event was personnel error.

The licensee's corrective actions included:

- All available licensed operators have been trained on the requirements for a Halon containment boundary breach.

- Security supervisory personnel, including the shift security supervisors, were trained regarding fire doors that are part of a Halon containment boundary and the required actions when they are found impaired.

- The fire protection impairment permit (Form FC-1142) will be revised by April 1, 1993, to document specifically whether the impairment will affect a Halon containment boundary.

The inspector reviewed documentation for the completion of the corrective actions taken by the licensee. Based on the review performed by the inspector, it appeared that the licensee had taken and plans to take appropriate actions to preclude repetition of this event.

6.6 <u>(Closed) Licensee Event Report 93-001: Failure to Satisfy Surveillance Requirement for Boric Acid Tank Level Check</u>

This licensee event report documented the licensee's failure to satisfy surveillance requirements for the boric acid tank level check. On January 8, 1993, while reviewing maintenance activities to be performed on Boric Acid Storage Tank B level indication, the licensee identified that the boric acid storage tank level surveillance was not being properly performed. The licensee performed the surveillance test by comparing two level indications (local and remote) that originated from the same sensor. This contradicted surveillance requirements which called for comparing two independent sensors.

The licensee determined that the root cause of the event was an inadequate surveillance test to monitor the boric acid storage tank levels. A review of plant records showed that the original 1973 surveillance test incorrectly identified the local and remote indications from the level bubblers as meeting the intent of comparing independent indications.

The licensee's corrective actions include:

- Temporary Modification 93-005 installed a temporary sight glass level indication on each boric acid storage tank and Surveillance Test OP-ST-SHIFT-0001, "Operations Technical Specification Required Shift Surveillance," was revised to address comparing sight glass and bubbler readings.

- Surveillance Test OP-ST-SHIFT-0001 has been reviewed and determined to adequately meet the intent of associated Technical Specification requirements.

- Modification MR-FC-93-001 has been initiated to split the current single annunciator associated with both boric acid storage tank float switches (LAS-260 and LAS-253). The modification will provide each boric acid storage tank with an individual annunciator window. Necessary procedure changes, including a change to Surveillance Test OP-ST-SHIFT-0001 to specify LAS-260 and LAS-253 as the independent sensors for comparison to the local and remote level indicators (LIT/LIA-261 and LIT/LIA-253) for the boric acid storage tanks, will be completed when the modification is accepted for operation. This modification will be installed during the outage that is scheduled to begin in April 1993.

The inspector reviewed documentation for the completion of the corrective actions taken by the licensee. Based on the review performed by the inspector, it appeared the licensee had taken and plans to take appropriate actions to preclude repetition of this event.

7 FOLLOWUP (92701)

7.1 (Closed) Inspection Followup Item 285/9036-01: Seismic Qualification of the Auxiliary Steam Piping in the Diesel Generator Rooms

This item involves a concern identified with respect to the auxiliary steam piping, used for space heating, in the diesel generator rooms not being able to withstand a design basis earthquake. It was noted that failure of the piping during a seismic event may adversely impact the operability of the emergency diesel generators.

The licensee performed a walkdown of the piping in the diesel generator rooms and confirmed that the piping could withstand a design basis earthquake, as installed. If a seismic event were to occur, the piping from the auxiliary boiler, which supplies steam for the system, would most likely fail since the piping outside safety-related areas is not seismically supported. The failure of the piping would eliminate the supply of steam to the diesel rooms; therefore, even if a failure of the piping in the rooms were to occur, no steam would be released into the rooms.

During review of this issue, the licensee noted that a high energy line break analysis had not been performed for the auxiliary steam piping. At the time of discovery, the steam supply had already been secured as a result of a previously identified condition. See the discussion on Licensee Event Report 91-010 in Section 6.2 of this report.

An analysis was performed by the licensee upon discovery that a line break review had not been performed. As a result of the analysis, the licensee determined that the condition was of minor safety significance for the following reasons:

- If a through-wall crack developed and steam filled the diesel generator rooms, the fire alarms would activate and notify the operations staff of an impending problem. The diesel generator rooms are connected via an opening in the wall between the diesel generators. A fire damper is installed in the wall and contains a fusible link. In the event of a steam line break, the temperature in the room would not elevate to a point that would melt the fusible link; therefore, the equipment in both diesel generator rooms would potentially be susceptible to a steam environment.

- The environmental conditions would reach limiting values in about 5 minutes in one diesel generator room and in about 50 minutes in the other diesel generator room. This would allow more than ample time for the onshift operators to assess the condition and to take actions to secure the steam supply prior to the environmental conditions reaching limits in both rooms.

- The ventilation systems in both rooms, once the diesel started, would evacuate the steam from the room and eliminate the potential for the environment in the rooms from reaching any limiting condition.

The licensee installed permanent modifications to eliminate the potential vulnerability by removing the fire damper in the wall between the diesel generator rooms and installing a fire door, which is shut during plant operation.

7.2 (Closed) Inspection Followup Item 285/9123-01: Long-Term Corrective Action to Increase the Emergency Diesel Generators Fuel Oil Supply

This item resulted from the licensee's discovery that the onsite supply of fuel oil for the emergency diesel generators would be sufficient for only 3.5 days. The Technical Specifications require a 7-day supply of fuel oil. The difference in the amount of fuel oil available was attributed to plant electrical loads that had been added since initial construction. The licensee had not accounted for additional loading on the emergency diesel generators, which increased the fuel oil usage.

To address this issue, the licensee provided the NRC with a short-term action plan for reestablishing the 7-day storage requirement. The NRC reviewed and concurred with these actions. The licensee plan stated that the actions listed below would be taken:

- Within 10 hours of the starting of the emergency diesel generators during a design basis accident, fuel oil would be ordered by the emergency response organization and arrangement would be made for continuous delivery of fuel oil to the site.

- In the event fuel oil cannot be delivered to the site because of weather conditions, fuel oil will be transferred from the 18,000-gallon auxiliary boiler fuel oil storage tank to the diesel fuel oil storage tank. This evolution would be performed by installation of a temporary modification. The transfer of the fuel oil would be accomplished by using the fuel oil pump on the third auxiliary feedwater pump. The development of a long-term plan to address this issue was an inspection followup item.

The licensee, in a letter dated July 1, 1992, proposed a permanent solution to the fuel oil supply problem. The licensee stated that, based upon a recent engineering analysis, current fuel oil capacity was sufficient to run Diesel Generator 1 for 4.9 days or Diesel Generator 2 for 5.2 days. This analysis took credit for postaccident load reductions. The proposed long-term plan was to utilize the short-term actions with some enhancements. The proposed enhancements are listed below:

- The time required to connect the two fuel oil storage tanks will be reduced by the addition of valves and piping to the auxiliary boiler fuel oil transfer system.

- Sufficient hose for the fuel oil transfer system will be dedicated, tagged, and stored in an appropriate area. A preventive maintenance task will be performed on this hose.

- A portable pump and hose will be dedicated as a backup transfer pump. A preventive maintenance task will be performed on this pump and hose.

- Appropriate procedure revisions will be completed and new procedures will be implemented as necessary for use of the transfer system.

- The auxiliary boiler storage tank will be upgraded to the same classification as the emergency diesel storage tank. This will include all the same quality assurance requirements, inspections, and administrative controls currently in effect for the emergency diesel storage tank.

- System walkdowns and training for appropriate personnel will be performed periodically.

- Technical Specifications and Updated Safety Analysis Report revisions will be submitted to clarify the crediting of the auxiliary boiler storage tank as part of the onsite emergency diesel generator fuel oil storage capacity.

This proposal was submitted to the Office of Nuclear Reactor Regulation and is currently under discussion with the licensee. This inspection followup item is closed based on the licensee's submittal.

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

*R. Andrews, Division Manager, Nuclear Services
J. Bobba. Maintenance Supervisor
*J. Chase, Manager, Fort Calhoun Station
*G. Cook, Supervisor, Station Licensing
*S. Gambhir, Division Manager, Production Engineering
*J. Gasper, Manager, Training
*W. Gates, Vice President, Nuclear
*R. Jaworski, Manager, Station Engineering
*L. Kusek, Manager, Nuclear Safety Review Group
D. Lippy, Licensing Engineer
*T. Patterson, Division Manager, Nuclear Operations
*R. Phelps, Manager, Design Engineering
*J. Sefick, Manager, Security Services
*C. Simmons, Station Licensing Engineer
*R. Short, Manager, Nuclear Licensing and Industry Affairs
J. Tills, Operations Supervisor

*Denotes 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 March 16, 1993. 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.