APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-285/92-15 Operating License: DPR-40

Docket: 50-285

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: July 19 through August 29, 1992

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

Approved: Elmo E. Collins for 9/16/92
P. H. Harrell, Chief, Project Section C Date
Division of Reactor Projects

Inspection Summary

Areas Inspected: Routine, unannounced inspection of onsite followup of events, operational safety verification, safety system walkdown, maintenance and surveillance observations, Temporary Instruction 2515/115, "Verification of Plant Records," and onsite followup of licensee event reports.

Results:

- Licensee actions following the discovery of an overheated power supply cable to a 125-Vdc distribution bus demonstrated a high level of concern with regard to safe plant operations. The licensee provided training to operations personnel on the simulator on what would have occurred in the plant if the cable had failed (paragraph 2.4).

- Radiological personnel support of the activities related to the testing and removal of the pressurizer code safety valves was very good (paragraph 3.3).

- Walkdowns of the containment spray system, engineered safeguard controls, and component cooling water system verified that these systems were in an operable status (paragraph 4.4).

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- Maintenance and surveillance activities were found to be well coordinated, with good procedural compliance, and good attention to personnel safety (paragraph 5.1).

Summary of Inspection Findings:

- Licensee Event Reports 91-006 and 91-023 were closed (paragraph 8).

Attachment:

- Attachment - Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

At the beginning of this inspection period, the Fort Calhoun Station had just ended its forced outage that began on July 3, 1992. The outage was the result of a reactor trip followed by a loss-of-coolant event. Plant heat up was initiated on July 20 and power ascension was continued until July 26 when the plant was administratively maintained at 90 percent power due to the licensee's concerns about its capability of meeting the requirements of Technical Specification 2.10.4(b), which concerned the availability of the incore alarms used for monitoring linear heat rate. On July 30, following a reevaluation of its concerns, licensee management gave permission for the operators to increase power to 100 percent. The plant operated essentially at 100 percent power until August 5, when the licensee began reducing power to 1 percent to replace an overheated power supply cable to 125-Vdc distribution Bus AI-41A. Upon completion of this effort, the licensee began increasing power again on August 6, and reached 100 percent power on August 7. On August 22, the Fort Calhoun Station experienced a reactor trip on thermal margin/low pressure with a premature lifting of Pressurizer Code Safety Valve RC-142. The Fort Calhoun Station remained shut down throughout the remainder of this inspection period to perform inspection and testing on both pressurizer code safety valves.

2 ONSITE RESPONSE TO EVENTS (93702)

2.1 Failure to Comply with Linear Heat Rate Technical Specification

On July 17, 1992, while reviewing Technical Specification 2.10.4(1)(b), the licensee identified a potential for previous Technical Specification violations involving linear heat rate monitoring requirements. The potential violations involved instances when the plant computer incore detector alarms were inoperable and conditions specified in Technical Specification 2.10.4(1)(b) might not have been satisfied. The review identified linear heat rate uncertainties and allowances of 11.8 percent, based on several factors referred to in the Technical Specification that had not been applied prior to June 24. It was determined that on May 15, May 22, May 29, and June 24, a condition in Technical Specification 2.10.4(1) that required power to be reduced to the limits of core operating limits, unless measured peak linear heat rate prior to the incore detector alarm outage was not greater than 90 percent of the allowable peak linear heat rate, may have been violated.

Following an initial review of this event, the licensee determined that the impact on the safe operation of the plant was minimal. Linear heat rate is used to provide information on core performance and fuel management and does not provide any automatic protective function. Data for these events indicated that the peak linear heat rate, before and after alarm inoperability, did not exceed the Technical Specification allowable peak linear heat rate. The licensee presently is applying the uncertainties and allowances of 11.8 percent to the peak linear heat rate, while awaiting a Technical Specification interpretation to be developed by its nuclear engineering department, to define the appropriate application of uncertainties/allowances to peak linear heat rate when operating under Technical Specification 2.10.4(1)(b)(i).

The inspectors will perform further review of this event during routine review of Licensee t Report 285/92-024.

2.2 Inadvertent Start of Emergency Diesel Generator 2

On July 23, 1992, while performing a surveillance test on Emergency Diesel Generator 2, an operator inadvertently started the diesel from the local control panel. As part of the surveillance test, the operator took the engine control switch to the local position. This caused an anticipated alarm at the local panel, which needed to be acknowledged at the panel. However, instead of pushing the acknowledge button, the operator pushed the start button. The diesel subsequently started and increased to idle speed, as expected. The operator notified the control room and the diesel was shut down.

The licensee determined the primary cause to be operator error. However, the licensee determined a lack of human factors consideration in the layout of the local control panel as another cause of the event. The safety significance of this event was minimal since Emergency Diesel Generator 1 was operable and in the emergency mode.

The inspectors will perform further review of this event during routine review of Licensee Event Report 285/92-025.

2.3 Nonconservative Incore Detector Alarm Limits for Monitoring Peak Linear Heat Rate

On July 23, 1992, the licensee discovered that the incore neutron flux monitoring system alarm limits were set nonconservatively. The limits had been calculated based on a peak linear heat rate limit of 15.22 kW/ft instead of the appropriate limit of 13.8 kW/ft. As a result, the required alarms would not have been received at the appropriate peak linear heat rate value. Therefore, the incore detector alarms would not have served their monitoring function as specified in Technical Specification 2.10.4(1). Thus, the limiting conditions for operation associated with inoperable incore detector alarms were not satisfied from the beginning of Cycle 14 (May 3 to July 23).

As a result, the licensee took the following immediate actions:

- The peak linear heat rate in the Combustion Engineering Core Operating Report was corrected to reflect the appropriate core operating limits report limit.

- The incore alarm limits were recalculated and installed in the emergency response facilities computer system.

- The peak linear heat rate limit in the mini-Combustion Engineering Core Operating Report program was changed to the appropriate core operating limits report limit. The mini-Combustion Engineering Core Operating Report program provides much of the same information as the Combustion Engineering Core Operating Report but is displayed on the emergency response facility computer display screen.

- The axial shape selection system program was verified to contain the correct peak linear heat rate alarm limits consistent with the Technical Specification/core operating limits report value.

The safety significance of this event was determined to be minimal based on the availability of the peak linear heat rate alarm capability from an alternate source (i.e., the axial shape selection system program). The use of the axial shape selection system program permitted on-line monitoring and alarm capacity for ensuring that the peak linear heat rate limit was not exceeded. In addition, a review of Combustion Engineering Core Operating Report data showed that the measured value of peak linear heat rate, with uncertainties/allowances, did not exceed the core operating limits report peak linear heat rate limit referenced by Technical Specification 2.10.4(1) during the time in question.

An investigation of the incorrect peak linear heat rate value determined that the incorrect value was used in the Cycle 14 Combustion Engineering Core Operating Report coefficient generation analysis performed by a contractor. It was found that the contractor's verification of their analysis failed to detect the error and that the contractor's review/verification procedure was inadequate. In addition contributing factors identified inadequately defined interface between the licensee and the contractor, lack of ready availability of certain design information to contractor engineers performing the analysis, and the licensee did not review the contractor's Combustion Engineering Core Operating Report analysis (review of such an analysis is optional under current licensee procedures).

The following are additional actions the licensee is taking:

- The contractor will be provided additional controlled copies of the Technical Specification and the core operating limits report by October 15. Updates will be provided for these documents consistent with procedures for controlled documents. This will minimize the probability of using outdated operating limits and system configurations in analyses performed for the licensee.

- The nuclear engineering department will perform reviews of future reload associated analyses performed by the contractor until December 31, 1993. At that time a determination will be made as to whether continuing reviews will be required. The monitoring of reviews and successful compliance with review process improvements will be the responsibility of the quality improvement team.

The inspectors will perform further review of this event during routine review of Licensee Event Report 285/92-026.

2.4 Overheated Power Supply Cable to a 125-Vdc Distribution Bus

On August 5, 1992, the licensee went from Mode 1 (power) to Mode 2 (hot standby) due to an overheated power supply cable to 125-Vdc distribution Bus AI-41A. The overheated cable is located in the control room.

The overheated cable was discovered by an electrical maintenance technician, who noted the odor of hot insulation while performing a fuse replacement in the back of the same cabinet. Further inspection revealed that the 125-Vdc power supply cable to Bus AI-41A had charred insulation at the switch connection to the bus. The licensee, using an infrared thermography camera, discovered an overheated condition on the cable at the switch connection. The licensee made the decision to place the plant in hot standby to work on the cable. The licensee was concerned that work on the cable and bus could result in the loss of power to Bus AI-41A, which would affect main feedwater, air-operated valves, and the fast transfer capability of the 4160-Vac busses.

The inspectors maintained continuous coverage in the control room during the shutdown. No problems were encountered. Prior to shutting down, the licensee took several actions, which included:

- Having the training department run a simulator scenario for a loss of Bus AI-41A.
- Briefing the oncoming crew and requiring them to practice this scenario on the simulator prior to relieving the onshift crew.
- Using the infrared thermography camera on other control room cabinets to determine if any other connections had unusually high temperatures. None were detected.

The licensee determined that the cause of this event was a failure internal to the associated switch, which raised the resistance levels at the switch connection with the cable. The licensee plans to send the switch to the Wyle Laboratory for a thorough inspection to determine the actual cause of the failure. A replacement switch was not available, thus a jumper wire was installed across the switch as a temporary modification. This work was performed using Temporary Modification 92-63 and Maintenance Work Order 923363. The damaged switch did not provide any automatic protective function, but served only to deenergize the bus prior to performing work on the bus.

2.5 Chemical and Volume Control System Relief Valve Outside of Design Basis

On August 17, 1992, the licensee reported that an overpressure protection device in the chemical and volume control system was installed in a configuration outside of the design basis. The overpressure protection device is Check Valve CH-202 which is installed in the bypass line for Valve HCV-238. Valves HCV-238 and HCV-239 are the two isolation valves for the charging system flowpath to the reactor coolant system.

The design function of the check valve is to provide overpressure protection for the charging side of the regenerative heat exchanger. Check Valve CH-202 is the only check valve installed for this purpose. However, if the in-line isolation valve (HCV-247) between Valve CH-202 and the regenerative heat exchanger were to be closed, then thermal expansion in the heat exchanger could not be relieved. Isolation Valve HCV-247 is normally open and fails open upon loss of power. Thus, the main concern would be for an inadvertent closure of the valve by operators. The licensee's immediate corrective action was to caution tag the control switch for Valve HCV-247 to alleviate any immediate safety concern. The licensee is considering other long-term corrective actions like a keylock switch on the control room panel.

The inspector will perform further review of this event during routine review of Licensee Event Report 285/92-027.

2.6 Reactor Trip and Premature Lifting of Pressurizer Safety Valve

On August 22, 1992, at 1:52 a.m., the Fort Calhoun Station experienced a reactor trip due to reactor coolant system pressure reaching the thermal margin/low pressure trip setpoint.

The event was initiated when a malfunction in the turbine electrohydraulic control system caused the four turbine control valves to go from the 40 to 22 percent open position, resulting in a partial loss-of-load condition. The closure of the control valves did not result in turbine trip since the modification installed during the recent forced outage was not designed to actuate until one valve was within 1/4-inch of its fully closed position. Thus, the partial loss-of-load, without a turbine trip, resulted in increased reactor coolant system temperature and pressure. The pressure reached 2397 psia when one out of the four high pressure trip channels actuated. The high pressure trip setpoint is 2400 psia and requires two out of four trip signals to initiate a plant trip. Before another high pressure channel trip was received Pressurizer Code Safety Valve RC-142 lifted prematurely at 2397 psia. The lift setpoint for Valve RC-142 was set at 2500 ± 25 psia. Thus, the decreasing reactor coolant system pressure reached the thermal margin/low pressure trip setpoint of approximately 2000 psia.

Valve RC-142 shut at approximately 1715 psia, which was within the design blowdown of the valve. Control room indication of tailpipe temperature confirmed that the valve was fully shut and not leaking.

All major equipment was operable and all safety systems functioned, as expected. The only equipment problems experienced were with the electrohydraulic control system and Pressurizer Code Safety Valve RC-142. Region IV sent three inspectors to the Fort Calhoun Station on August 24, to investigate the premature lifting of Valve RC-142 and the failure of the electrohydraulic control system. The licensee maintained the plant in hot shutdown conditions to test both pressurizer code safety valves (RC-141 and RC-142) while in place. The results were inconclusive and the decision was made to go to cold shutdown, remove Valves RC-141 and RC-142, and send them to the Wyle Laboratory in Huntsville, Alabama, for inspection and testing. This was completed and the valves were reinstalled on September 1, 1992. The results of the inspection by the Region IV inspectors are documented in NRC Inspection Report 50-285/92-21.

2.7 Conclusions

The licensee's overall response to these events was found to be very good (i.e., taking prompt and conservative corrective actions). The electrical maintenance technician who discovered the overheated cable on Bus AI-41A demonstrated a good questioning attitude and may have prevented an unnecessary challenge to the plant's safety systems. Licensee management's decision to brief the oncoming crew of operators during this event demonstrated a good safety attitude and good use of the simulator.

3 OPERATIONAL SAFETY VERIFICATION (71707)

3.1 Routine Control Room Observations

The inspectors observed operational activities throughout this inspection period to verify 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. The inspectors noted that providing a shift supervisor briefing to all oncoming crew members in the operator's loft, prior to the shift turnover, was very good. This was a recent enhancement to the shift turnover process. 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.

3.2 Plant Power Reduction

One of the nonroutine events experienced by the licensee operators occurred on August 5, 1992, when the Fort Calhoun Station underwent a reduction in power operation from Mode 1 to Mode 2 due to an overheated power supply cable to the 125-Vdc distribution Bus AI-41A. As described in Section 2.4 of this report, the efforts taken by the operations personnel to minimize the chance of an inadvertent reactor trip was excellent and proactive.

3.3 Testing of Valves RC-141 and -142

On August 24, 1992, the resident inspectors and Region IV inspectors witnessed the testing of Valve RC-142 and the control room monitoring of activities. The inspector noted that a licensed operator in the control room was in constant communication with personnel performing the test in containment. The operator was monitoring plant conditions to ensure that, after each lift of the relief valve, a loss of coolant did not exist. The testing was delayed due to the time required for setting up and calibrating the test equipment. Thus, the actual test was not begun until a few minutes before 11 p.m. The control room shift turnover normally occurs between 11:00 - 11:30 p.m. The inspector noted that the licensee delayed the shift turnover until the test was completed at approximately midnight. This prevented changing personnel during the performance of the test.

Personnel performing the test inside containment maintained good radiological protection practices, due in part to the briefing that was provided by radiological protection personnel prior to the test teams entry into containment. Despite the severe ambient temperatures encountered, the test was performed with good communication flow between the test team and the control room.

The licensee's decision to maintain the same operators until the test was completed was appropriate.

3.4 Replacement of a Reactor Coolant Pump Seal

On August 30, 1992, the licensee entered mid-loop operations to replace the seal package on Reactor Coolant Pump RC-3D. Prior to this effort, all three shift crews were briefed by management on the pertinent procedures, per the guidance in the licensee's Standing Order G-92, "Conduct of Infrequently Performed Procedures."

During this briefing, the operators were reminded on the need to limit site activity that may adversely impact site power and shutdown cooling capabilities. As a result, the switchyard was locked and access to the switchgear rooms was limited to those personnel that were required to have access, such as plant security and the turbine building operators. In addition, the operators were advised that in the event that shutdown cooling was lost, they had less than 132 minutes to regain it.

The licensee used several methods to monitor the reactor coolant system level. These included the narrow- and wide-range level indicators located on the control room panel and the site glass located in containment, which licensee personnel monitored every 15 minutes. In addition, the licensee had a meter/recorder that measured the amperage of the shutdown cooling pump. This meter/recorder would provide an alarm if certain changes in the pump amperage occurred, which was an indication that the pump may be air binding. This indication would reveal that the reactor coolant system level may have dropped below the desired level and that the pump suction line may be uncovered.

At approximately 3 p.m., on August 30, the licensee began inventory reduction. The licensee reached mid-loop conditions at approximately 2 a.m., on August 31, when the licensee began the repair efforts on the reactor coolant pump. The licensee completed this effort at 6 a.m.

3.5 Plant Tours

The inspectors toured various areas of the plant to verify that proper housekeeping was being maintained. Housekeeping was found to be excellent. Various valve and switch positions were verified for the current plant conditions. Personnel were observed obeying rules for escorts and visitors, and entry and exits into and out of vital areas.

3.6 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.

Radiological protection personnel performance during those activities described in Section 3.3 of this report was very good. Such activities included providing radiological protection briefings prior to any effort where the opportunity for personnel exposure was high or if personnel were to be working in a hazardous or stressful environment. In addition, when licensee personnel were in containment to test the pressurizer code safety valves, radiological protection personnel maintained a close watch of the test team to verify that the heat encountered (approximately 117[illegible] was not having an adverse effect.

3.7 Security Program Observations

The inspectors observed security personnel perform their duties of personnel and package search. Vehicles were properly authorized and controlled or escorted within the protected area. Designated vehicles parked and unattended within the protected area were found to be locked and the keys removed. The inspectors routinely toured the protected area perimeter and found it maintained at an excellent level. Also noted was that proper compensatory measures were taken when a security barrier was inoperable.

3.8 Technical Specification Waiver of Compliance

On July 21, 1992, the licensee requested a one-time waiver of compliance from the provisions of Technical Specification 3.17(3)(iii)3 regarding inservice surveillance of steam generator tubes. The applicable Technical Specification section requires that unscheduled inservice inspections be performed on each steam generator during a shutdown following a loss-of-coolant accident that resulted in the actuation of the engineered safeguards equipment. The July 4, 1992, loss-of-coolant event resulted in a partial engineered safeguard features actuation.

The NRC granted the waiver on July 21. The waiver was granted based on the licensee's determination that the event produced no significant transient on the steam generator tubes which could have caused any existing indications in the tubes to propagate or any new degradation to occur. This waiver was valid only for the restart from that event.

3.9 Conclusions

Operations, radiological protection, and security personnel were observed to be performing their duties in an excellent manner. In addition, the overall physical condition of the plant was maintained in good condition.

4 ENGINEERED SAFETY FEATURE SYSTEM WALKDOWN (71710)

The inspectors walked down accessible portions of the following systems to verify operability, as determined by verification of selected valve and switch positions.

4.1 Containment Spray - Normal Operation

The valve positions were verified using Operating Instruction OI-CS-1, "Containment Spray - Normal Operation." The inspector noted that all valves inspected were in their proper position and that all locked valves were labeled requiring shift supervisor's approval prior to manipulation.

4.2 Engineered Safeguard Controls - Normal Operation

The switch positions were verified using Operating Instruction OI-ES-1, "Engineered Safeguard Controls - Normal Operation." The inspector noted that all switches were in their proper positions.

4.3 Component Cooling Water - Normal Operation

The valve locations and system configuration were verified using the appropriate piping and instrumentation drawings. The valve positions indicated in these drawings were further verified by comparing them to Procedure OI-CC-1, "Component Cooling System Normal Operation." In addition, the inspector performed a walkdown of control room panels CB-1/2/3, CB-4 and AI-45, and verified that the valve positions for accessible and nonaccessible valves, as indicated by the associated lights and switches, were in the proper position per Procedure OI-CC-1.

The overall condition of the system piping and valves was good. No valve packing leaks or other notable valve damage, such as bent valve stems, missing handwheels or improper labeling, was identified.

4.4 Conclusions

The inspectors concluded, based on verification of system status, that the containment spray system, engineered safeguard controls, and the component cooling water system were capable of performing their intended safety functions.

5 MAINTENANCE OBSERVATIONS (62703)

The inspectors observed selected station maintenance activities on safety- and nonsafety-related systems and components.

5.1 Replace Power Supply Cable to A 125-Vdc Distribution Bus

On August 5, 1992, the inspector witnessed the removal of an overheated power supply cable and the main feeder switch (AI-41A-MAIN) to 125-Vdc distribution Bus AI-41A, followed by the installation of a jumper wire as a temporary modification. The purpose for this effort is described in Section 2.4 of this report. The work was performed in accordance with Maintenance Work Order 923363 and Temporary Modification 92-63. The maintenance work order was approved, as noted by the appropriate signatures.

The licensee began this effort by reducing power to approximately 1 percent power to minimize the effects on the plant should the power supply cable fail before the maintenance work was completed. The panels to Bus AI-41A were removed and a fan was installed to help dissipate the heat generated by the cable and reduce the cable temperature.

To remove the power supply cable and the associated main feeder switch, the licensee first installed a temporary feed line from Bus AI-41B to Bus AI-41A. The purpose of this line was to bypass the main switch and the power supply cable to maintain Bus AI-41A energized when the normal power to the bus was removed. This allowed electrical maintenance personnel to disconnect the normal power supply and remove the power supply cable and the associated main switch. The temporary modification installed consisted of a jumper wire, which bypassed the location of Main Switch AI-41A-MAIN. This was installed because the licensee was unable to locate an onsite replacement for the switch. It was noted that the damaged switch did not provide any automatic protective function.

The licensee personnel performed this effort in a safe and efficient manner by maintaining good communication and by taking precautionary measures to minimize personnel injury. Some of these measures included using rubber mats and insulated gloves and by isolating the work zone to minimize traffic in the work area. In addition, management personnel oversight was apparent due to the significance of this effort.

5.2 Charging Pump Packing Cooling Pump CH-1A-1 Replacement

On August 6, 1992, the inspector witnessed portions of the licensee effort to replace the charging pump packing cooling Pump CH-1A-1 with a modified pump. The modified pump contained an oil seal to prevent Xenon gas, originating through the charging pump plunger packing, from leaking past the pump shaft and elevating radiation exposure levels in the charging pump room. This work was part of an overall effort to improve the performance of all three charging pumps and to extend the life of the charging pump plunger packing.

This effort was accomplished per Modification Request MR-FC-90-046. The inspector verified that all required equipment was properly tagged out-of-service and that the work instructions were adequate to control this activity. In addition, it was observed that proper care was used in performance of the task by personnel maintaining good radiological protection practices. The modification package had been reviewed and approved prior to its use as indicated by the appropriate signatures.

5.3 Conclusions

The licensee's maintenance activities were found to be well coordinated with good procedural compliance. Maintenance personnel adhered to good radiological protection practices, and took appropriate precautions to improve personnel safety.

6 SURVEILLANCE OBSERVATIONS (61726)

6.1 Safety Injection System Category A and B Valves

On August 20, 1992, the inspector witnessed portions of the performance of Surveillance Test OP-ST-SI-3001, "Safety Injection System Category A and B Valve Exercise Test." This test is performed quarterly to satisfy the requirements of Technical Specification 3.3(1)a by ensuring the integrity of the reactor coolant system and other components subject to inspection and testing according to ASME XI Boiler and Pressure Vessel Code.

The test was performed by licensed and nonlicensed operators and instrumentation and control technicians. Good coordination and communication was noted among all involved. The inspector witnessed the prejob briefing and noted good communication of responsibilities by the reactor operator in charge of the test.

The surveillance was performed efficiently with good attention-to-detail. The portions of the surveillance test witnessed by the inspector were performed in accordance with the procedure. At Step 1, Attachment 3, of the surveillance procedure, it was noted by the operator that two redundant valves had their location on the control room panels reversed. Although this would not have made a difference in the performance of the test, the operator suspended the test and notified both the shift supervisor and the shift technical advisor. The shift technical advisor immediately initiated the required paperwork to revise the procedure. The test was then resumed with no other procedure anomalies noted.

During the performance of the test, all the valves tested had stroke times that did not require action. However, one valve was tested in an alert range and appropriately noted in the remarks column of the test procedure.

The inspector reviewed the completed and approved copy of the surveillance test and found all required signatures were present.

6.2 Conclusions

The licensee was noted to be properly implementing their surveillance program with very good attention-to-detail.

7 REVIEW OF TEMPORARY INSTRUCTION 2525/115, "VERIFICATION OF PLANT RECORDS" (2515/115)

This temporary instruction provides guidance for evaluating each licensee's ability to obtain accurate and complete log readings from either licensed or nonlicensed operators.

7.1 Discussion

The inspector reviewed a selected number of turbine building (Form FC-78), auxiliary building (Form FC-143), and control room logs, and made a comparative analysis between these logs and security room entry records. This review encompassed 19 licensed and nonlicensed operators and covered the 24 hour periods of May 15, June 28, June 30, and July 1, 1992. Areas covered in this review included rooms that contained plant safety equipment, such as the diesel generator rooms and the switchgear rooms.

7.2 Conclusions

No errors or discrepancies were noted in this review. All logs coincided with personnel entries into the associated rooms.

8 ONSITE REVIEW OF LICENSEE EVENT REPORTS (92700)

8.1 (Closed) Licensee Event Report 285/91-006: Failure to Establish Compensatory Firewatches

This licensee event report described conditions that occurred on March 5, 1991, in which the licensee failed to establish a continuous fire watch on two inoperable fire barriers following the inadvertent inoperability of the associated fire detection zone.

The root cause of this event was determined to be inadequate administrative controls to ensure that the modification, which installed a new alarm panel (XL-3/AI-56), was properly reflected in the operating procedures and the training lesson plans. This resulted in an inadequate understanding by operations personnel of how the modified detection system worked, thus causing them to fail to recognize system impairment (i.e., operations personnel did not realize the entire detection zone was inoperable when one of the detectors in the zone failed to reset following a false alarm). Thus, a continuous fire watch, as required by Technical Specification 2.19(1), was not established.

The following corrective actions were implemented by the licensee to preclude recurrence of this event:

- Procedure OI-FP-4, "Fire and Smoke System and Alarm Procedure," was revised to make it clear that, if a detector on a zone controlled by Fire Alarm Panel AI-54A/B is in alarm, the remainder of the detectors in that zone are inoperable or, if a detection zone controlled by AI-54A/B indicated a trouble condition, the detection zone should be considered inoperable until the cause of the trouble is determined. In addition, Procedures ARP-CB-20/A14, "Annunciator Response Procedure A14 Control Room Annunciator A14," and ARP-AI-54A/A54, "Annunciator Response Procedure A54 Control Room Annunciator A54, Fire Detection Panel," were similarly revised.

- The licensee revised their lesson plans to clearly define the operation of detection zones controlled by AI-54A/B and the interface between the two fire alarm panels.

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.

8.2 <u>(Closed) Licensee Event Report 285/91-023: Failure to Comply with Technical Specification 2.10.4(1)(b)(iii)</u>

This licensee event report documented the failure of operations personnel to maintain power levels constant while the emergency response facilities computer system was shutdown for maintenance purposes. On October 22, 1991, while the emergency response facilities computer system was inoperable, the licensee allowed power levels to increase from 98.2 percent to 99.3 percent contrary to the requirements of Technical Specification 2.10.4(1)(6)(iii).

Following a human performance enhancement system evaluation, the cause of the event was determined to be human error. The primary causal factors included failures in communication and work practices. The operators were aware of the need to operate within the conditions imposed by Technical Specification 2.10.4(1)(b). They failed, however, to perform a full review of the Technical Specification and all of its required actions prior to entering the limiting conditions for operation.

The licensee's corrective actions included the development of an abnormal operating procedure, which provides the operations crews with specific guidance and corrective actions if the emergency response facilities computer system is off line due to system failures or for maintenance. In addition, this incident was reviewed in the operator requalification training and a procedural change to Procedure OP-4, "Load Change and Normal Power Operation," was made to include a precaution on the requirements of Technical Specification 2.10.4 with the emergency response facility computer system inoperable.

The inspectors reviewed the actions taken by the licensee and noted no problems. It appeared that the actions will prevent recurrence of this event.

ATTACHMENT

1. PERSONS CONTACTED

1.1 Licensee Personnel

*J. Chase, Assistant Manager, Fort Calhoun Station
*S. Gambhir, Division Manager, Production Engineering
*R. Jaworski, Manager, Station Engineering
*L. Kusek, Manager, Nuclear Safety Review Group
*T. Patterson, Manager, Fort Calhoun Station
*R. Phelps, Manager, Design Engineering
*C. Simmons, Station Licensing Engineer
*R. Short, Manager, Nuclear Licensing and Industry Affairs

*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 September 1, 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.