APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-285/92-09 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: March 15 through April 25, 1992

Inspectors: R. Mullikin, Senior Resident Inspector
R. Azua, Resident Inspector
E. Collins, Project Engineer, Project Section C

Approved: P H Harrell 5-13-92
P. H. Harrell, Chief, Project Section C Date

Inspection Summary

Inspection Conducted March 15 through April 25, 1992 (Report 50-285/92-09)

Areas Inspected: Routine, unannounced inspection of licensee event report (LER) followup, onsite followup of events, operational safety verification, and maintenance, surveillance, and refueling activities.

Results:

- Prompt response by the operators, to the event that resulted in a short loss of shutdown cooling, was excellent. However, the event could have been avoided with an appropriate procedure and better control over outage activities. As a result, a violation was identified (paragraph 4.e).
- The licensee's design basis reconstitution program continued to identify potential safety concerns (paragraphs 4.b, -f, and -g).
- The licensee's actions previously implemented for an identified concern with electrical switches appeared to be insufficient (paragraph 4.h).

- Improvements in the licensee's foreign material exclusion program may be needed after a flashlight end cap was dropped into the refueling cavity (paragraph 4.i).

- Management oversight during this outage was considered excellent (paragraph 5.e).

- Operator performance with regard to surveillance procedural compliance and initiative in withdrawing from the test when confronted with complications was excellent (paragraph 7).

DETAILS

1. Persons Contacted

*R. Andrews, Division Manager, Nuclear Services
J. Chase, Outage Manager, Fort Calhoun Station
*G. Cook, Supervisor, Station Licensing
M. Frans, Supervisor, Systems Engineering
*S. Gambhir, Division Manager, Production Engineering
*J. Gasper, Manager, Training
*W. Gates, Division Manager, Nuclear Operations
*R. Jaworski, Manager, Station Engineering
*L. Kusek, Manager, Nuclear Safety Review Group
*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. Tills, Outage Manager, Fort Calhoun Station
D. Trausch, Supervisor, Operations

The inspectors also contacted additional personnel during this inspection period.

*Denotes attendance at the monthly exit interview on April 28, 1992.

2. Plant Status

The Fort Calhoun Station was in its 13th refueling outage during this entire inspection period. Fuel reload was commenced on March 29, 1992, and completed on March 31. At the end of this inspection period, the plant was heating up in preparation for restart.

3. LER Followup (92700)

a. (Closed) LER 90-005: Spent Fuel Pool Ventilation System Outside Design Basis

This LER documented that VA-66, the spent fuel pool area charcoal filtration unit, was outside its design basis due to insufficient air flow into the unit from the spent fuel pool area. A fuel handling accident assumes that all activity released would be filtered through VA-66. Licensee testing was unable to confirm that the air flow in the spent fuel pool area supported this assumption.

The licensee evaluated the condition in Safety Analysis for Operability (SAO) 90-002. NRC Inspection Report 50-285/90-13 reviewed the immediate safety impact of this condition and left the issue open pending review by the Office of Nuclear Reactor Regulation.

The NRC reviewed the existing configuration and evaluated the consequences of a fuel handling accident. The results of this review were documented in a letter to the licensee, dated August 14, 1991. The NRC concluded that the existing condition was acceptable based on the potential offsite doses being within the standard review plan guidelines.

b. (Closed) LER 90-025: Containment Cooling Water (CCW) Systems Outside Design Basis

This LER documented the licensee's determination that the normally-closed raw water (RW) valves, which interface with the CCW system, would fail open on a loss of instrument air, rendering the CCW system inoperable. Also, other air-operated valves would fail open, reducing the flow of cooling water to the containment coolers in the design basis accident. The licensee also identified design weaknesses in the RW system that limited its ability to act as a backup cooling water supply to the containment coolers and the possibility that one containment spray pump could be placed in the configuration of supplying two spray headers. This could put the pump in a runout condition and possibly reduce its service life or render it inoperable.

The licensee concluded that one containment spray pump with a two header configuration was adequate to limit containment pressure in the design basis accident and that additional modifications to the CCW/shutdown cooling heat exchanger isolation valves and CCW isolation valves to the containment coolers were planned for the 1993 refueling outage.

The licensee implemented a modification using nitrogen to keep one containment spray valve closed to prevent the pump from operating in a runout condition. In addition, the licensee locally jacked the CCW/RW interface valves closed to prevent them from opening on a loss of instrument air.

NRC Inspection Report 50-285/90-42 reviewed the conditions identified by the licensee and the short-term corrective actions. The review concluded that the as-found containment cooling configurations were adequate to control containment pressure following the design basis accident and that there would be time for manual actions to restore the CCW functions.

Based on the short-term corrective actions that were implemented and the NRC review of the safety significance of these conditions in NRC Inspection Report 50-285/90-42, this LER is closed.

c. (Closed) LER 91-008: Inappropriate Surveillance Requirements for Reactor Protection System Level 1 Bistables

This LER documented the licensee's voluntary report of the discovery that the four Level 1 bistables on the power range nuclear instrument units of the reactor protection system had not been tested monthly. The monthly testing requirement was per a Technical Specification interpretation and was later determined to be invalid because of the system design, which prevented testing of the bistables wh the plant was at power. The licensee also determined that the four bistables had not been calibrated since initial plant operation. During plant startup, the bistables enable the loss-of load and axial-power distribution trips and disable the high power rate-of-change trip when reactor power increases above 15 percent.

The licensee committed to calibrate the Level 1 bistables prior to the next reactor startup and develop a surveillance test to functionally test the bistables prior to each startup. In addition, the licensee planned to revise the Technical Specification interpretation to reflect the correct surveillance test for the bistables and implement a Technical Specification verification action plan to compare the existing surveillance procedures to the Technical Specifications to ensure that each required surveillance has a corresponding procedure. The planned completion date for the program is July 1, 1992.

The inspector reviewed the calibration data sheets for the Level 1 bistables and verified that the bistables had been calibrated. The inspector also reviewed Surveillance Procedure IC-ST-RPS-0043, "Power Range Safety Channel Level 1 Bistable and Turbine Loss of Load Function Test," and verified that procedural requirements to test the Level 1 bistables had been met.

The inspector concluded that the safety significance of this condition was minimal since, at the time of discovery, the Level 1 bistables were in the correct state for the plant conditions. In addition, the Level 1 bistables had been verified during the last plant startup to be in the correct state prior to putting the main generator on line at approximately 12 percent power, and the Level 1 bistables had been verified during the last plant shutdown to be in the correct state at 10 percent power.

d. (Closed) LER 91-012: Emergency Diesel Generator Autostart Due to Loss of Transformer T1A-4

This LER documented an unplanned engineered safeguards features actuation when an electrician bumped an uncovered relay inside a transformer control cabinet, generating a transformer lockout and starting Emergency Diesel Generator 2.

The licensee identified that the causes for the actuation were that electricians were unfamiliar with the color coding for the transformer control cabinet wiring and that the relay was uncovered, making it vulnerable to external manipulations.

The licensee's corrective actions included briefing electricians on the need for caution to be used inside transformer control cabinets, installing labels to more clearly identify components in the transformer control cabinets, adding the event to Electrical Training Lesson Plan 12-61-01, "Codes, Standards, and Procedures," and initiating a design change to replace the uncovered relays with covered relays.

The inspector reviewed documentation for the completion of the corrective actions. The design change to replace the relays was planned for the current refueling outage but had not yet been implemented. Based upon the completed corrective actions and the commitment to replace the relays with covered relays, this LER is closed.

e. (Closed) LER 91-031: Personnel Air Lock (PAL) Door Connections Outside of Design Basis

This event concerned the leak rate connections for the PAL doors, which were determined not to be seismically qualified. The PAL consists of a cylindrical steel barrel with a bulkhead welded to each end. Each bulkhead has a door with two seals each. The seals on the doors are tested according to 10 CFR Part 50, Appendix J, Type-B, leak rate testing at 5 psig after each opening or daily, whichever is less frequent. The entire PAL assembly is tested at 6-month intervals, at the maximum containment design pressure of 60 psig. The design utilized test caps located on each bulkhead to test the seals and existing gauge taps on the outer bulkhead to run the sensing line from the inner door to the test panel. This design was intended to allow testing with both PAL doors closed. In addition, there was a pipe stub and valve for pressurizing the entire assembly to the 60 psig test pressure.

The licensee determined that there was a lack of documentation attesting to the seismic qualification of the test connections, thus, the testing appartus was outside the design basis. Immediate corrective action was to danger tag closed the PAL door and remove the unqualified components from the test connections and cap the lines. However, this required testing the inner door from within the barrel area with the outer door open. The licensee subsequently installed qualified tubing and valves, which would allow testing the doors with them closed. The inspector verified that this permanent modification was installed. In addition, the licensee will submit changes to the Updated Safety Analysis Report and the Technical Specifications reflecting this modification.

f. (Closed) LER 92-001: Unmonitored Release on Loss of the 13.8-kV Power Supply

This event involved the loss of power to the sample pump for the radioactive waste processing building exhaust stack radiation monitors (RM-041, -042, and -043). The loss of the sample pump was caused by a momentary loss of 13.8-kV power. The circuitry for the sample pump did not allow for automatic restart of the pump once power was restored; thus, operator action was required to start the pump. During the 5 minutes that the sample pump was inoperable, the exhaust fans were still in operation. This constituted an unmonitored release, but the licensee concluded that no release limits were exceeded.

The licensee installed Temporary Modification 92-04, which was a sample pump switch changeout to allow the pump to automatically restart when power is restored to the pump. In addition, the licensee initiated long-term corrective actions to add control room annunciation upon loss of sample pump power and evaluate the design of the sample pump and exhaust fan control circuitry. The actions taken and proposed satisfy the concerns related to this event.

g. (Closed) LER 92-002: Compromise of Containment Integrity Due to PAL Door Seal Leakage

The compromise of containment integrity occurred when the inner PAL door failed its leak rate test with the outer door open. The failure of the inner PAL door to pass its surveillance test was due to a piece of corrosion product that was lodged between the inner door seals. The surface corrosion was caused by condensate from CCW piping dripping onto the inner PAL door bulkhead structure and upper latch bolt bracket.

The testing of the inner PAL door, with the outer door open, was due to the test line traveling to the outside of the outer door not being seismically installed. This was documented in LER 91-031 (see paragraph 3.e). The licensee's initial corrective action was to remove the test line, but this required testing of the inner door from within the personnel access area.

The licensee's subsequent corrective actions included installing a seismically-qualified test line so both doors could be tested when both doors are closed. The inspector verified that the new test line was installed and that corrosion products from the inner door were removed. The licensee will evaluate the possibility of installing a deflector shield above the inner PAL door to prevent accumulation of water on the door assembly. The licensee's actions were sufficient to satisfy this concern.

h. (Closed) LER 92-003: Missed Fire Watch Due to Personnel Error

This event concerned a fire zone alarm being acknowledged by a control room operator but not reset. During welding activities in the area of the CCW heat exchangers, the Zone 2 fire alarm actuated on several occasions. The control room operators acknowledged and successfully reset the alarm except for one instance. The licensee determined that the Zone 2 alarm was inoperable for approximately 5 hours. Zone 2 included several fire detectors and all the detectors in that zone were inoperable contrary to Technical Specification requirements, which required an hourly fire watch. However, the licensee determined that only one room in Zone 2 was without an already established fire watch. That room was the lower mechanical penetration room and the licensee determined that a fire would not have gone undetected for very long.

The licensee's corrective actions included revising Standing Order M-9, "Fire Protection During Flame Cutting, Grinding and Welding Operations," and Form FC-18, "Flame Cutting and Welding Permit." The revisions required that fire detectors in the immediate area of welding activities be identified and compensatory measures be provided. The inspector verified that these procedure changes were completed.

i. (Closed) LER 92-004: Main Steam Safety Valves Outside Setpoint Acceptance Criteria

This event involved the surveillance testing on the 10 main steam safety valves. Five of the valves were found to be outside of their respective acceptance criteria for lift setpoints. Technical Specification 2.1.6(3) requires, at power operation, that eight safety valves be operable. The main steam safety valves are considered operable if they lift within plus or minus 1 percent of the nominal nameplate setpoint values. The five failed valves tested with a variation of +2.3 to +3.2 percent of the nameplate value.

The licensee had submitted a license amendment request to the NRC, on June 28, 1991, to change the tolerance from plus or minus 1 percent to +3.0 to -2.0 percent. Based upon this criteria, only one valve would have been declared inoperable and the Technical Specification requirement would not have been violated. However, this amendment had not been granted at the time of the event.

The licensee's immediate corrective action was to recalibrate and test the five failed valves. The licensee's actions are sufficient to satisfy the concerns in this event.

4. Onsite Followup of Events (93702)

a. Radioactive Waste Building Radiation Monitors Removed from Service

On March 17, 1992, Radiation Monitors RM-041 and -042, for the radioactive waste building, were removed from service to change the cartridge filter. The licensee initially determined this event to be a violation of Technical Specification 2.9.1.h.(i), which states that the particulate and iodine activity monitors may be inoperable provided that samples are continuously collected, as required in Table 3-12. The sampling frequency in Table 3-12 requires a weekly charcoal sample. This initial determination was made as a conservative measure; however, further discussion by the Plant Review Committee concluded that Technical Specification requirements had been met. Thus, the licensee decided that an LER was not required.

The inspectors reviewed the licensee's conclusion on LER reportability. It was concluded that the Technical Specification requirement for a continuous collection, per Table 3-12, meant continuous weekly charcoal sampling. This was satisfied by the replacement of the charcoal filter. Thus, the licensee's decision to not report this event appeared to be correct.

b. Containment Isolation Valve Outside of Design Basis

On March 20, 1992, the licensee determined that the service air isolation valves associated with containment Penetration M-74 were outside of the design basis. It was determined that the manual isolation valve (CA-555) inside of containment was a normally open valve and that the normally closed valves downstream were not seismic. Thus, in normal operation, the licensee could take credit for only the automatic valve (HCV-1749) outside of containment. The Updated Safety Analysis Report indicates that, generally, two containment isolation valves are provided for a containment atmosphere exposed system. However the licensee was taking credit for only one valve (HCV-1749), based on line pressure being greater than containment design pressure at all normal and postulated accident conditions, due to the air compressors maintaining pressure in the line. The licensee determined that credit could not be taken for the air compressors since these are not automatically loaded onto the emergency diesel generators following the loss of offsite power.

The licensee's immediate corrective actions included removing Valve CA-555 and installing a blank flange in its place. In addition, the licensee reviewed all other containment penetration valve arrangements and found no other problems. These actions eliminated any immediate safety concerns.

The inspectors will perform further review of this event during routine review of LER 92-011.

c. Reactor Protection System Trip Setpoint Greater Than Allowed by the Technical Specifications

On March 25, 1992, the licensee determined that the asymmetrical steam generator transient reactor trip setpoint was greater than that allowed by Technical Specification 1.3, Table 1-1. The Technical Specification allowed trip setpoint is less than or equal to 135 psid.

The trip unit has pretrip and trip setpoints that correspond to values of 100 and 135 psid, respectively. The licensee determined that all four trip channels were higher than allowed by the Technical Specifications. Based upon the surveillance test used, the maximum acceptable voltage would correspond to a differential pressure of 138.5 psid. The as-found voltage for all four channels corresponded to 136.125 psid, which was greater than allowed. The licensee concluded that this event was not safety significant with the higher than allowed setpoints. Immediate corrective actions included revising the subject procedure and reviewing other procedures for similar problems.

The inspectors will perform further review of this event during routine review of LER 92-012.

d. Radiation Monitor Sample Valves Inadvertently Closed

On April 8, 1992, the licensee reported that sample isolation valves for plant stack Radiation Monitors RM-050 and -051 had closed while a containment purge was in progress. The valves went closed when instrumentation and control personnel, working in a control room panel, accidentally lifted the leads to the sample valves. The operators were alerted by the technicians that the leads had been lifted. Within approximately 15 minutes, the leads were relanded and the purge secured. The valves were subsequently tested and declared operable.

The inspectors will perform further review of this event during routine review of LER 92-013.

e. Loss of Shutdown Cooling

On April 12, 1992, the licensee reported an approximate 7-minute loss of shutdown cooling. At the time of this event, the licensee was performing flow testing on two high pressure safety injection pumps. Due to maintenance work being performed, three 480-volt busses were being supplied through one 480-volt breaker.

The three 480-volt busses (1B3A, 1B3A-4A, and 1B4A) being tied together was an abnormal electrical lineup. Normal supply to Busses 1B3A and 1B3A-4A is through Breaker 1B3A and normal supply to Bus 1B4A is through Breaker 1B4A.

After the two high pressure safety injection pumps were secured following performance of the test, the 480-volt supply breaker (1B3A) to the three busses tripped, apparently on thermal overload. This resulted in the loss of power to 120-Vac Instrument Bus 2. This bus was being supplied, at the time, from a bypass transformer that was itself supplied from one of the affected 480-volt busses (1B4A). Since the newly installed Emergency Battery 2 was in recharge, the inverter was not available to supply Instrument Bus 2. This resulted in a loss of power to the shutdown cooling flow control valve (FCV-326) controller and shutdown cooling flow indication.

The control room operators recognized that the shutdown cooling system flow control valve had failed open and that shutdown cooling flow control indication was lost. The operators decided to secure the running low pressure safety injection pump (SI-1A) to prevent a possible runout condition. Within approximately 7 minutes, the operators determined that the 480-volt breaker had tripped, apparently due to the test in progress; reset the breaker; and restored shutdown cooling. The reactor coolant temperature rose approximately 6°F during the time shutdown cooling was lost.

On April 23 the licensee completed Investigation Report SRG-92-287 for the event. The Nuclear Safety Review Group identified the following root causes:

- A failure to have a policy during shutdown to specify a normal electrical lineup for performing tests. The surveillance procedure used for this test did not specify what lineup would be required.

- Surveillance Procedure OP-ST-SI-3007, "High Pressure Safety Injection System Pump and Check Valve Test," did not adequately list the initial plant conditions required to perform the test (required electrical lineup).

- The outage control center was bypassed for this test and the test was performed 1 day earlier than scheduled. It was noted that routine surveillance tests did not normally get approval by the outage control center.

- Ineffective monitoring of work in progress by the outage scheduling group. The abnormal electrical lineup should have precluded the test from being performed.

- A perceived importance attached to the performance of this test at that time, with minimal regard for the status of supporting systems.

On April 13 the licensee tested the breaker and determined that the breaker tripped due to the thermal overload that was experienced due to the test configuration. This surveillance test had not previously been performed with three 480-volt busses supplied through one breaker and full flow testing of the high pressure safety injection pumps.

The abnormal electrical lineup that was in effect at the time of the surveillance test should have precluded the test from being performed. The outage control center was formed to help prevent events such as this. The bypassing of the outage control center for certain tests increased the probability of this event. Regardless, Procedure OP-ST-SI-3007 should have given proper initial conditions (required electrical lineup) to safely perform the test, and thus, the procedure was inadequate. This is a violation of NRC requirements. (285/9209-01)

f. <u>125-Vdc Breaker Coordination Study</u>

On April 16, 1992, the licensee reported the results of its 125-Vdc breaker/fuse coordination study. This study was conducted as part of the licensee's design basis reconstitution evaluation of the electrical power distribution overcurrent trip capability.

The study determined that a coordination problem existed between the 1600-amp output fuse on both emergency batteries and several molded-case circuit breakers on loads supplied by the two dc busses. The concern was that, due to the tripping characteristics of the fuse and breakers, a possibility existed that the 1600-amp battery supply fuse to the dc bus could blow before the individual dc bus load breaker tripped.

The circuit breakers in question were those that protect the following equipment:

- Battery Chargers 1 and 2

- Inverters 1 and 2
- Crossconnect breakers to Battery Charger 3
- Turbine emergency bearing lube oil pump (LO-4)

The licensee replaced the 1600-amp fuses with those that would coordinate with the load breakers.

The inspectors will perform further review of this event during routine review of LER 92-010.

g. Potential for Inadequate Suction Head During Containment Spray Pump Recirculation Phase

On April 20, 1992, the licensee reported the results of the design basis reconstitution of the containment spray system. The results indicated that the containment spray pumps would not have adequate net positive suction head during the recirculation phase of safety injection. The licensee stated that the original design analysis indicated that each containment spray pump would have a flow rate of 2000 gpm in the recirculation phase. The new analysis showed that the actual flow would be approximately 3000 gpm, which would result in approximately 4 feet less net positive suction head than required.

On April 25 the licensee approved SAO 92-02, "Inadequate Containment Spray Pump Net Positive Suction Head." The SAO stated that, although the current licensing basis did not allow for subcooling in calculating net positive suction head, normal engineering practice allows it. The licensee calculated that more than 19 feet of subcooling would be available under all accident conditions using subcooling.

The licensee performed a 10 CFR Part 50.59 safety evaluation and determined that no unreviewed safety question existed. The SAO will be in effect until a change is made to the Fort Calhoun Station licensing basis or until the next refueling outage, when a modification to the containment spray pumps would be made.

The inspectors will perform further review of this event during routine review of LER 92-016.

h. Cracking of Cam Followers on General Electric Control Switches

On April 20, 1992, the licensee reported cracked plastic cam followers on 4160-volt breaker switches. The licensee discovered the cracking after a failure of RW Pump AC-10A during testing. The licensee decided to perform a 100 percent inspection of all

General Electric Type-SBM control switches based upon NRC Information Notice 80-13, which detailed cracking of cam followers made from a material called Lexan. This type of switch was installed at the Fort Calhoun Station during construction and were made from either Lexan or Delrim. Cams made from Delrim have not had a history of cracking. The inspection of the 4160-volt switchgear found 55 out of a total of 191 switches made from Lexan. Out of the total of 55 switches, 40 had some type of degradation. After bounding the problem in the 4160-volt switchgear, the licensee expanded the inspection scope to the control room, which was the other location of these type of switches.

The licensee, based upon acceptance criteria from General Electric, identified 17 switches that needed replacement. Three of these were in the control room and the rest were in the switchgear rooms. None of the switches would have prevented the equipment from being operated either remotely or locally. The licensee prioritized the switch replacements by those that were needed prior to heatup, prior to exceeding a reactor coolant temperature of 300°F, and prior to criticality. In addition, five switches were identified that could be replaced while the plant was on line. The licensee did not have enough spare switches onsite and was able to purchase a supply from General Electric. At the end of this inspection period, the switch replacement was progressing satisfactorily.

The inspector questioned the licensee on what action had been taken in regard to Information Notice 80-13. The following information was provided by the licensee, giving the chronological history of the switch problem:

- In 1976 the licensee received a General Electric service information letter, which described the fracture of the Lexan cam followers due to exposure to hydrocarbons during manufacture. General Electric recommended replacing switches, used in safety-related equipment, that had been manufactured between 1972 through 1976. The licensee determined that none of their switches were manufactured during this period.

- A supplement to the service information letter was issued in 1976, which stated that switch replacement was not recommended unless an inspection revealed severe cracking.

- In 1979 another supplement to the service information letter reminded users to inspect switches periodically until determining that no deterioration of the switch function had occurred.

- Information Notice 80-13 was issued, which alerted utilities to switch failures that had occurred at other plants.

- The licensee removed, in 1982, one switch for inspection and found that 50 percent of the cam followers on this switch had cracking similar to that described in Information Notice 80-13. A detailed inspection program was recommended using a fiber-optic scope.

- In 1984 an inspection approach, using a fiber-optic scope, was abandoned due to insufficient clarity. The licensee staff recommended that consideration be given to replacing switches in safety-related application in lieu of periodic inspections.

- During the 1985 refueling outage, the licensee replaced approximately 30 of the 90 safety-related switches in the control room. Nine of the replaced switches were completely dismantled for inspection. Some cracking was discovered, but the licensee concluded that the cracking was stress related rather than from exposure to hydrocarbons. No cam failures were discovered. Based upon this, the licensee decided to discontinue the changeout of the switches.

The licensee's decision to suspend plant heatup and replace the unacceptable switches was conservative. This demonstrated an awareness to safety versus meeting schedules. However, the the licensee's decision, in 1985, to not continue a General Electric recommendation for periodic inspections of the switches did not appear conservative based on inspection results during this outage.

The inspectors will perform further review of this event during routine review of LER 92-017.

i. <u>Flashlight Cap in the Reactor Vessel</u>

On April 2, 1992, at approximately 12:45 a.m., during the performance and preparation of incore removal from the upper guide structure platform, a flashlight that was being used on the job fell apart. The technician, to whom the flashlight was secured via a lanyard, was able to catch the batteries that fell out but was unable to grab hold of the end cap and the spring attached to it. Both cap and spring fell through a hole in the platform into the cavity.

The licensee began an immediate verification of tool accountability and initiated a search to locate the flashlight end cap. Incident Report No. 920235 was written on this event.

Initial inspection around the vessel was unsuccessful in locating the end cap. Thus, it was determined that it had fallen into the reactor vessel and, as a result, the licensee developed a systematic inspection plan.

The licensee located the end cap and spring in a control element assembly shroud within the upper guide structure. The licensee retrieved both the cap and the spring and initiated a review of their foreign materials exclusion program.

Conclusions

Prompt response, by operations personnel, to the event that resulted in a short loss of shutdown cooling was excellent. However, the event could have been avoided with an appropriate procedure and better control over outage activities.

The licensee's design basis reconstitution program continued to identify potential safety concerns.

The licensee's prior actions on an identified concern with electrical switches was insufficient.

5. Operational Safety Verification (71707)

a. Routine Control Room Observations

The inspectors observed operational and outage activities throughout this inspection period. Proper control room staffing was maintained and control room professionalism and decorum were observed. Discussions with operators determined that they were cognizant of plant status and were aware of plant activities that could affect plant safety. The inspector observed selected shift turnover meetings and noted that excellent transfer of information concerning plant status and planned evolutions occurred between the offgoing and the oncoming operators.

b. Plant Tours

Plant housekeeping was found to be maintained as work activities decreased and plant personnel endeavored to return to areas that were the focus of maintenance activities to their preoutage conditions.

On April 16, 1992, the inspector noted that two cells (Cells 36 and 37) of the newly installed Emergency Battery 2 were darker in color than the adjoining cells. An inspection of all the other cells for both batteries revealed no similar color differences. The inspector notified the licensee of the finding. Station engineering responded that the color difference was noted upon receipt inspection and that a discussion with the battery manufacturer (C&D) had occurred. The licensee was assured by the manufacturer that the darker color was not a sign of copper contamination and that all of the cells will turn darker after a few months of use. The reason given for these two cells was that they may have received a longer charge at the factory. Based upon the conversation with C&D, the licensee concluded that these two cells were good.

During routine tours of the plant, the inspectors noted the presence of the occupational safety coordinators on a routine basis. There was a safety coordinator present on all shifts. Their continuing presence and noted interaction with plant workers demonstrated a licensee commitment to overall plant safety.

c. Radiological Protection (RP) Program Observations

The inspectors verified that selected activities of the licensee's RP program were implemented in conformance with facility policies, procedures, and regulatory requirements. Radiation and/or contaminated areas were properly posted and controlled. Health physics (HP) personnel were observed to be touring work areas to ensure that proper radiological protection practices and radiological control requirements were properly implemented.

On April 13, 1992, as part of the internal exposure control program, the licensee performed a random whole body count on a site employee. During this process, the licensee identified a small amount of internal contamination of Cesium-137. Although the levels of contamination, less than 1 MPC-hour, did not rise to the level for reportability, the HP technician noted that any indication is uncommon, thus RP management was notified. RP management interviewed the employee to determine when and where this contamination may have occurred. The employee identified that, on February 28, 1992, three employees, including herself, were found to have facial contamination when exiting the radiation control area. RP management looked for the personnel contamination report for this event but discovered that none had been written. As a result, Radiation Occurrence Report No. 92-09 was initiated.

The findings of the radiation occurrence report and its associated root cause analysis were the subject of a separate NRC inspection and is documented in NRC Inspection Report 50-285/92-07.

d. Security Program Observations

The inspectors observed that personnel, packages, and vehicles were properly searched before entering the protected area. It was noted that guards were posted when vital area doors were open for plant activities. Isolation zones were found to be free of transient material.

e. Observation of Management Activities

Throughout this inspection period, management involvement in outage activities continued to be very visible. Management personnel gave briefings prior to the initiation of infrequently performed procedures per the guidance in Standing Order G-92, "Conduct of Infrequently Performed Procedures." The inspector witnessed a member of licensee management provide such a briefing prior to the performance of the 10-year inservice inspection hydrostatic test of the Class 3 components of the component cooling water system. Management's commitment to plant safety was apparent. Management was noted touring the plant spaces on a routine basis. The operations superintendent was noted to be routinely in the control room interacting with operators.

Conclusions

Operational personnel performed their duties in a professional manner during the plant shutdown and during the preparation for returning the plant to power. Security personnel were identified as being knowledgeable of their responsibilities and performed their duties as required. Management oversight during this outage was found to be excellent.

6. Maintenance Observations (62703)

a. Machining of the Access Port Seal Bore for Valve SI-194

On March 25, 1992, the inspector observed the preparation for and machining of the Loop 2A low pressure safety injection header check valve (SI-194). The purpose of this effort was to correct the out-of-round condition on the body pressure sealing area for Valve SI-194. The personnel involved in this effort adhered to the procedure requirements and maintained good communication and cooperation with HP personnel. Adherence to RP principles was found to be good (i.e., preplanning and prestaging of equipment was notable in minimizing personnel stay times in the area). Proper use of respiratory equipment was noted. In addition, steps were taken by the personnel involved to prevent the dissemination of the debris developed during this effort. Management oversight of this effort was apparent.

b. Reinstallation of Containment Isolation Valve CA-555

On March 25, 1992, the inspector observed the reinstallation a containment isolation valve (CA-555). The valve had been removed for repairs following its failure to pass a Type-C leak rate test. Due to the valve's location high above the ground, a work platform was erected for this effort, which provided a safe and sturdy work station for the personnel involved.

The associated maintenance work order (921385) was reviewed and approved as noted by the appropriate signatures, and the activity was found to be within the skills of the trade of the personnel involved. Radiological controls were found to be appropriate and were properly implemented.

c. Replacement of 4160-Volt Switch

On April 21, 1992, the inspector witnessed the replacement of a switch in the switchgear cabinet for 4160-volt Breaker 1A4-10 under Maintenance Work Order 921872. This is the supply breaker to 4160/480-volt Transformer T1B-4A. The replacement was performed due to cracking cam followers for certain switches, as documented in paragraph 4.h. The inspector noted that the work was performed in a careful and controlled manner. This and the other switch replacements were delaying plant heatup. However, there was no indication that the craftspersons felt rushed to complete the job. The inspector reviewed the completed work order and found no problems.

Conclusions

Maintenance was found to be performed in a coordinated, controlled manner, with adherence to procedures. Adherence to RP principles was good, especially in the areas of preplanning and prestaging of equipment.

7. Surveillance Observation (61726)

On April 16, 1992, the inspector witnessed the performance of the equipment qualification testing, as specified in Procedure SE-EQT-RW-0001, "Determination of Component Cooling Water Heat Exchanger Raw Water Outlet Valve Throttle Setting." The purpose of this test was to set the upper limit on RW flow through a single CCW heat exchanger, at approximately 3500 gpm, to preclude RW pump operation in the runout condition, with one pump and two heat exchangers in operation. This test was part of the acceptance testing for Modification No. FC-90-026.

The inspector monitored this effort for procedural compliance and noted that all limiting conditions for operation were met and that precautions were observed. Good communication between the control room operator running the test and the personnel stationed locally at the valve was noted. At one point during the test, while the operator was attempting to isolate three of the heat exchangers, an RW outlet valve (HCV-2882B) for CCW Heat Exchanger AC-1B failed to close. Unable, at the time, to determine the cause, the operator backed out of the test, returning the equipment back to its original condition. The test was not resumed until the cause of the problem was identified (the air solenoid for Valve HCV-2882B was found to be sticking). The operator was found to be knowledgeable of his responsibilities and of the purpose of the test.

One observation, made by the inspector during this effort, was that the operator performing this test was also involved in supporting efforts on another surveillance. At one point during the test, the operator was observed talking over the Gaitronics to two separate people simultaneously, on the same channel, concerning the two separate tests and signing off steps on both procedures. The inspector verified that no errors were made by the operator during both these efforts. The inspector questioned the wisdom of this practice, for it may lead to confusion and increase the chances of error on the part of the people involved. The plant manager was alerted and stated that th[illegible] normal practice.

Conclusions

Good communication was noted between operations and [illegible] personnel. Operator performance with regard to procedural compliance and initiative in withdrawing from the test when confronted with complications was considered to be excellent.

8. Refueling Activities (60710)

On March 29 and 30, 1992, the inspectors witnessed portions of the reload of fuel from the spent fuel pool to the reactor vessel. The licensee offloads the entire core during each refueling, which is 133 fuel assemblies. This reload included approximately one third new fuel.

The inspector noted that the portion of the reload witnessed was done in a careful manner. It was observed that the refueling cavity water was somewhat murky and visibility was less than when the core was offloaded. The underwater camera was not in use at the time; however, it could have been turned on if a problem arose. In the opinion of the inspector, water clarity was sufficient for camera use.

9. Summary of Open Items

The following is a synopsis of the status of all open items generated and closed in this inspection report.

- LERs 90-005, 90-025, 91-008, 91-012, 91-031, 92-001, 92-002, 92-003, and 92-004 were closed.

- Violation 285/9209-01 was opened.

10. Exit Interview

The inspectors met with Mr. W. G. Gates (Division Manager, Nuclear Operations) and other members of the licensee staff on April 28, 1992. The meeting attendees are listed in paragraph 1 of this inspection report. At this meeting, the inspectors summarized the scope of the inspection and the findings. During the exit meeting, the licensee did not identify as proprietary, any information provided to, or reviewed by, the inspectors.