APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-285/91-03 Operating License: DPR-40

Docket: 50-285

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

Facility Name: Fort Calhoun Station (FCS)

Inspection At: FCS, Blair, Nebraska

Inspection Conducted: January 16 through February 26, 1991

Inspectors: R. Mullikin, Senior Resident Inspector
T. Reis, Resident Inspector

Approved: [signature] 3-7-91
R. H. Harrell, Chief, Project Section C Date

Inspection Summary

Inspection Conducted January 16 through February 26, 1991 (Report 50-285/91-03)

Areas Inspected: Routine, unannounced inspection of onsite followup of events, operational safety verification, maintenance observations, safety-related system walkdown, licensee event report (LER) followup, review of previously identified inspection findings, and followup on Three Mile Island (TMI) items.

Results:

- The licensee actions were proactive the on-line leak repair of a flange on a primary code safety valve. Even though the leak rate was allowable by the Technical Specification (TS), plant management made the decision to stop the leakage before the leakage increased (paragraph 3.a).

- The licensee's design basis reconstitution program continued to provide benefits in that two issues (Penetration M-3 integrity and offsite power low signal setpoint error) were discovered (paragraphs 3.b and 3.c).

- Adequate implementation of the radiation protection and security programs was noted with the exception of a personnel contamination event that occurred on January 29, 1991 (paragraphs 4.c and 4.d).

- Housekeeping continued to be very good (paragraph 4.b).

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- Implementation of the maintenance program appeared to be adequate (paragraph 5).

- Safety-related system walkdowns indicated that selected systems were aligned for the applicable plant condition (paragraph 6).

DETAILS

1. Persons Contacted

M. Bare, System Engineer
*J. Chase, Manager, Nuclear Licensing and Indust.y Affairs
*J. Gasper, Manager, Training
*W. Gates, Division Manager, Nuclear Operations
*R. Jaworski, Manager, Station Engineering
*L. Kusek, Manager, Nuclear Safety Review Group
D. Lovett, Supervisor, Radiation Protection
D. Matthews, Supervisor, Station Licensing
*T. Matthews, Station Licensing Engineer
*W. Orr, Manager, Quality Assurance and Quality Control
T. Patterson, Manager, Fort Calhoun Station
*A. Richard, Assistant Manager, Fort Calhoun Station
*J. Sefick, Manager, Security Services
*R. Sexton, Supervisor, Radiation Health and Administration
C. Simmons, Station Licensing Engineer
F. Smith, Supervisor, Chemistry
T. Therkildsen, Supervisor, Nuclear Licensing
*S. Willrett, Manager, Nuclear Materials and Administration

NRC

L. Ricketson, Radiation Protection Specialist, Region IV

The inspectors also contacted additional personnel during this inspection period.

*Denotes attendance at the monthly exit interview.

2. Plant Status

The FCS operated at 100 percent power from the beginning of this inspection period until February 11, 1991, when a reduction in power to 75 percent was initiated.

On February 11 the licensee announced that the 1991 refueling outage, previously scheduled to begin on September 28, would be delayed until January 30, 1992. This delay was necessary due to unexpected forced outages that have occurred since the last refueling. To extend the fuel cycle to January 1992, the licensee decided to operate at 75 percent power until load demand warrants an increase. However, on February 22, the licensee decided to reduce power to 70 percent to conserve fuel and to eliminate oscillations on a main feedwater regulating valve. The plant remained at 70 percent power throughout the remainder of this inspection period.

3. Onsite Followup of Events (93702)

a. Primary Code Safety Valve F e e Leak

During startup from the recent forced outage that ended on January 14, 1991, the licensee noted that a flange on the piping to primary code safety valve (RC-142) was leaking. The startup was terminated and repairs were made to the flange during cold shutdown. When startup was resumed, the licensee noted that the flange connection was still leaking, but at a smaller rate. The licensee decided to resume the plant startup and monitor the leak during weekly containment entries.

On January 18 the licensee made a containment entry to inspect the leaking flange connection. The leak appeared to have increased from the time of plant startup. The licensee decided to leave the insulation off the flange to see if a temperature differential across the flange was the cause of the leak. Another entry was made on January 19 and, although the leak appeared to have decreased from the day before, it was still a concern. The decision was made to obtain the services of a contractor to inject sealant material into the flange.

On January 24 the plant review committee (PRC) met to review Temporary Modification 91-004, which included the contractor's engineering procedure and the licensee's engineering evaluation for the repair of the leaking flange. The inspector attended the PRC meeting and noted an exchange of ideas with emphasis on plant safety. Based upon the information obtained at the meeting, it was decided that the contractor's procedure needed to be upgraded to reflect the licensee's engineering analysis. Some of the changes identified were to limit the amount of sealant injected and to limit the pressure at which the material was to be injected. The PRC reconvened later the same day and the temporary modification package was approved. The inspector verified that a quorum of PRC members was present at the meeting.

On January 25 an attempt by the contractor to stop the leak was unsuccessful. See paragraph 5 for a description of this maintenance activity. A second attempt, on January 26, appeared to be successful.

Subsequently, weekly containment entries have been made to inspect the leak repair. It was noted that the flange was again leaking but much less than before the repair. However, when the leak rate was noticeably increasing, the licensee decided to have the contractor attempt another repair. On February [illegible] the contractor successfully repaired the leak. Subsequent flange inspections during this inspection period indicated no leakage.

The licensee still intends to perform a periodic surveillance of the flange to ensure that the integrity of the seal repair was maintained. A permanent fix will be made during the next extended cold shutdown.

b. Containment Penetration M-3 Integrity

On February 4, 1991, the licensee determined that a problem existed with containment Penetration M-3 (chemical and volume control system (CVCS) charging line). Previously, the NRC determined that Penetration M-3 did not require Type C testing per the requirements of 10 CFR 50, Appendix J, and was reflected as such in the TS. However, during the licensee's review of an open item identified in the design basis reconstitution program, it was determined that Penetration M-3 did require Type C testing.

A safety evaluation report (SER), issued by the NRC on January 10, 1986, stated that an exemption from Type C testing for Penetration M-3 was not needed because the charging pumps and system design prevented pressure from falling below the postaccident containment pressure of 2 psig. This penetration has a single check valve, CH-198, outside of containment. The original SER justification stated that the charging pumps, which automatically start and align suction to the boric acid storage tanks, provided a seal barrier against the escape of the containment atmosphere. After the boric acid storage tanks emptied, the charging pumps would be secured and the remaining head of water in the system would provide a seal against containment leakage.

However, based on revisions to the postaccident pressure analysis, the containment pressure was determined to be 20-40 psig instead of the original 2 psig. Revisions to this analysis were based upon containment spray nozzle blockage and containment cooling unit restrictions that the licensee had previously identified and reported to the NRC. After the charging pumps are secured, the calculated head in the CVCS system would be 6 psig. Thus, containment integrity through Penetration M-3 could no longer be assured.

The licensee declared Penetration M-3 operable based upon the existence of a check valve on each of the discharge lines of the three charging pumps. Although check valves cannot be used as a containment isolation valve, and these three valves had never been leak rate tested, the licensee had a high confidence in these valves. This confidence was based upon biweekly preventive maintenance (PM) performed on the nitrogen charged accumulator between the charging pump and the discharge check valve. Although this PM does not determine a leak rate, it would indicate whether or not the check valves would leak at full system pressure. However, this would not guarantee that the check valves would fully seat at the lower postaccident pressure of 20-40 psig. To account for this, the licensee issued Operations Memorandum 91-01 stating that, after the charging pumps are secured, the manual isolation valve on the discharge side of each check valve must be closed.

On February 5, 1991, a conference call was conducted between Region IV, the Office of Nuclear Reactor Regulation (NRR), and the licensee to determine if an immediate safety concern existed. During this call, the licensee responded that personnel would be available to shut the charging pump discharge valves, even with their other postaccident duties assigned. Also, the three manual isolation valves had been recently cycled so there was assurance that they could be physically closed. The licensee stated that once these valves are closed they would not be required to be opened again during this accident. The NRC also questioned whether the effects of leakage were considered on other systems connected to the line between Penetration M-3 and the manual isolation valves. The licensee responded that they had, and no problems were discovered. On February 6 the licensee issued Safety Analysis for Operability (SAO) 91-01 for Penetration M-3.

On February 7 a meeting was held in the Region IV office to discuss, among other topics, the operability of Penetration M-3. At this meeting, the licensee presented the staff a copy of SAO 91-01. The licensee stated that they would present to the staff, by February 22, the long-term corrective actions that would be taken to address this issue. It was agreed that the SAO would be formally docketed. This was done via a letter dated February 14.

On February 22 the licensee informed the inspector that, to resolve the Penetration M-3 long-term issue, procedure changes will be made. The changes will be designed to ensure that charging line pressure will not exceed containment pressure and will maintain the intent of the original SER. The licensee stated that changes to abnormal and emergency operating procedures will be completed by May 15. This position will be formally documented in LER 91-003.

Followup of this issue will be done during review of LER 91-003.

c. <u>Offsite Power Low Signal (OPLS) Setpoint Error</u>

On February 12, 1990, the licensee reported that a potential condition existed outside of the design basis in the station's degraded voltage protection system. In this system, an OPLS is received which sheds vital bus loads, ties in the emergency diesel generators (EDG), and resequences safety-related loads on the vital busses. To get an OPLS, there must be a safety injection actuation signal (SIAS) and a degraded voltage condition on the 4160-volt bus. The problem identified by the licensee was that the potential existed for the 161-kV offsite power system voltage to degrade to a level (above the OPLS setpoint) such that adequate voltage would not be present to ensure long-term operation of certain safety-related 480-volt equipment. The 161-kV supplies the 4160-volt bus, and the 4160-volt bus supplies the 480-volt bus. Analysis indicated that, at degraded bus voltages, the OPLS relays would maintain adequate voltage for all 4160-volt motors but would not protect the 480-volt motors from a reduced voltage condition.

The licensee issued, on February 12, 1991, Operations Memorandum 91-02 to institute immediate compensatory measures. These measures were to put a dedicated nonlicensed operator on each shift to monitor the 4160-volt bus voltages. If the voltages dropped below the new calculated setpoints and an SIAS was present, the operator would be required to actuate OPLS within 60 seconds, with the permission of the shift supervisor or senior reactor operator. In addition, this operator would be required to remain on shift until new setpoints were installed in the OPLS actuation relays. The new setpoints were installed on February 13, and the dedicated operators were relieved of their temporary duties.

The licensee's long-term corrective action will be the installation of a modification to incorporate automatic tripping of the feedwater and condensate pumps upon an SIAS. This will ensure that adequate voltages will be available for the 480-volt, safety-related motors. The licensee stated that this, in conjunction with the revised set of OPLS setpoints, will provide the final, long-term resolution to meet all design basis requirements for the degraded voltage protection system.

The inspectors will follow up on the licensee's long-term corrective action during the review of LER 91-004, which is being issued by the licensee to document the details of this problem.

Conclusions

The Penetration M-3 and OPLS issues were discovered as part of the licensee's resolution of open items generated during the design basis reconstitution program. These are examples of the benefits this program has gained for the licensee in increased awareness of safety-related issues.

4. Operational Safety Verification (71707)

a. Routine Control Room Observations

The inspectors observed operational activities throughout this inspection period to verify that adequate control room staffing was maintained, control room professionalism was adequate, and 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 status and the reasons for lit annunciators. Control room indications of various valve and breaker lineups were verified for current plant condition.

b. Plant Tours

On February 18, 1991, while confirming correct breaker positions on motor control centers, the inspector noted that position indicator bulbs for two pieces of equipment were apparently burned out. This was discussed with the shift supervisor and it was noted on his bulb replacement list. The inspector noted that the bulbs were replaced on the same day.

The inspectors routinely toured various areas of the plant to verify that proper housekeeping was being maintained. Generally, housekeeping was well maintained throughout the plant. Painting activities in the radiation controlled area were progressing and a marked improvement in appearance was noted.

c. Radiological Protection Program Observations

On January 25, 1991, the inspector made a containment entry at full power to witness the on-line leak repair of the primary code safety valve (RC-142) flange. See paragraph 5 for details.

The inspector attended the prejob briefing given by a radiation protection technician prior to entry. The briefing was comprehensive and easily understandable. Areas of high dose rates were pointed out and ALARA considerations were evident. The inspector noted that the technician was performing surveys while the work was being done to detect any changing conditions. He also prompted everyone involved to read their dosimeters regularly.

On February 1, 1991, the inspector was notified by the licensee of an incident which occurred on January 29. The incident involved the radiological contamination of eight persons due to the failure of polyethylene bags in which solid radiological waste was being transported.

As solid radiological waste was being transported from the drumming area in the auxiliary building to a loading van in the radwaste building, a bag containing a sharp vacuum filter tore, resulting in contamination of the auxiliary building and radwaste building floors, as well as eight personnel. The highest contamination level encountered from the spill was 35,000 dpm/100 cm^2 on the shoes of one of the workers.

The incident was documented in Radiological Occurrence Report 91-0010. In reviewing the licensee's report, it appeared that appropriate initial response, investigation, and proposed corrective actions were taken or proposed. The Region IV Division of Radiological Safety and Safeguards will review this incident during a future inspection.

d. Security Program Observations

The inspectors verified that selected activities of the licensee's security program were being adequately implemented.

On January 25, 1991, at approximately 9 p.m., the inspector observed that fog caused a reduction in the visibility within the protected area, causing the security cameras to be inadequate. The inspector observed that security guards were patrolling the protected area perimeter.

In addition, on February 18, 1991, the inspector toured the central alarm station and verified that cameras could adequately cover the protected area boundaries.

e. TS Interpretation of Channel for Core Exit Thermocouples

On December 11, 1990, the NRR responded to a Region IV request for a TS interpretation as to what constitutes a channel for the core exit thermocouples (CET). For the CETs at the FCS, there are two channels with each channel consisting of 14 CETs. There are seven CETs in each core quadrant, with four of them in one channel and the other three in the redundant channel.

Presently TS 2.21, Table 2-10 states, in part, that "With the number of OPERABLE Core Exit Thermocouples less than the four required by NUREG-0737, either restore to at least four OPERABLE channels within seven days of discovery of loss of operability, or prepare and submit a special report . . . ."

NRR stated that Table 2-10 needed to be revised to bring it into conformance with plant design and the intent of NUREG-0737. Therefore, the TS should be changed to state "With the number of OPERABLE Core Exit Thermocouples less than the four per core quadrant required by NUREG-0737, either restore at least four OPERABLE Core Exit Thermocouples per core quadrant within seven days of discovery of loss of operability, or prepare and submit a special report . . . ."

At the monthly exit meeting on February 26, 1991, the licensee committed to making the TS change. This will be an inspector followup item (IFI) (285/9103-01).

f. TS Interpretation of Onsite Presence of Various Individuals

In the December 11, 1990, letter discussed above, NRR responded to a request as to what constitutes "onsite duty" for the plant manager, operations personnel, and radiation protection operator/technician positions. NRR stated that the plant manager is considered to be absent from the plant when he/she leaves the site boundary (i.e., the owner-controlled property). As applied to the personnel specified in TS Section 5, it means that the individuals are physically inside the protected area unless job responsibilities, during normal or accident conditions, require them to perform duties outside the protected area but within the site boundary.

The licensee stated, at the monthly exit meeting, that this interpretation was consistent with the directions that they had provided to their personnel.

5. Maintenance Observations (62703)

a. On-line Leak Repair of Primary Code Safety Valve RC-142 Flange

On January 25, 1991, the inspector witnessed the initial attempt by a contractor to perform a leak repair of the flange between the pressurizer and Valve RC-142. The work was performed under Maintenance Work Order (MWO) 910296.

When the flange was removed during the recent outage, it was discovered that the misalignment of the tongue and groove assembly of the flange had occurred during the previous installation. To prevent this from happening again, the licensee split the flange spacer ring so that alignment could be visually inspected. After the tongue and groove were mated, the split ring could then be inserted and the flange bolted together. The spacer ring does not constitute a pressure barrier but only limits the amount of compression on the pressure boundary gasket.

The decision was made by the contractor to use the two splits (180 degrees apart) in the spacer ring for sealant injection points after tapping and threading in adaptor plugs. The steam leak was coming out of both splits. The adaptor plugs were successfully installed and the sealant material was injected in one side and then the other. However, this was not totally successful since a small leak was noticed coming from one of the eight flange stud holes. At that point, the decision was made to exit the containment and determine the next course of action.

On January 26 Revision 1 to Temporary Modification 91-004 was made and approved by the PRC. This revision allowed drilling a third hole through the spacer ring near the leaking stud hole to stop the leak. The licensee and contractor decided that the sealant material was probably setting up too quickly due to the heat in the piping and not spreading where it needed to. It was decided to thin the sealant before injecting and try to stop the leak using the existing holes before drilling a third hole. A second attempt at stopping the leak was made later that day. The sealant material was injected and the leak was successfully stopped.

b. Reactor Coolant Pump (RCP) RC-3A ressure Breakdown Device

On January 29, 1991, the inspec.ors witnessed the attempt to press apart the partially blocked first-stage, pressure breakdown device that was removed from RCP RC-3A. This blockage resulted in a plant shutdown on August 24, 1990.

During the initial investigation as to the cause of the blockage, the licensee discovered some metal balls at the outlet of the breakdown device. This was thought to be the remains of some missing lock wires that were used on the bolts that attached the lower-seal assembly to the middle-seal assembly. However, before these metals balls could be analyzed, they were discarded as solid waste. See NRC Inspection Report 50-285/90-38 for details.

The licensee determined that more physical evidence existed in the breakdown device. Thus, MWO 904454 was prepared to press apart the breakdown device from the lower seal and retrieve any evidence.

The licensee attempted to press apart the breakdown device while using liquid nitrogen to cool the device. This was unsuccessful, and the decision was made to heat the outer portion of the seal using a torch while pressing the breakdown device. This required the use of a tent around the work area and a different radiation work permit. On January 31 the breakdown device was successfully removed and one small metal ball was discovered.

The licensee decided to send the metal ball, along with a lock wire, to Combustion Engineering in Connecticut for analysis. The results of this analysis will be discussed in a future inspection report.

c. Replacement of EDG 1 Intake Damper Solenoids

On February 5, 1991, the inspector witnessed a portion of the replacement of the solenoid valves for the fresh air intake dampers for EDG 1. These solenoids were replaced as part of Modification Request MR-FC-89-085. No problems were noted during the inspector's observations.

Conclusions

The maintenance activities witnessed by the inspectors were performed in accordance with procedures and in a professional manner.

6. Safety-Related System Walkdown (71710)

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

- Diesel Generator No. 1 - Starting Air System, Checklist OI-DG-1-CL-A and Drawing B120F07001, Sheet 1, Revision 20
- Diesel Generator No. 1 - Fuel Oil System, Checklist OI-DG-1-CL-B and Drawing 11405-M-262, Sheet 1, Revision 33
- Diesel Generator No. 2 - Starting Air System, Checklist OI-DG-2-CL-A and Drawing B120F07001, Sheet 1, Revision 20
- Diesel Generator No. 2 - Fuel Oil System, Checklist OI-DG-2-CL-B and Drawing 11405-M-262, Sheet 1, Revision 33
- Plant Electrical Distribution (4160-Volt System), Checklist OI-EE-1-CL-A and Figure 8.1-1, Revision 51
- Plant Electrical Distribution (480-Volt System), Checklist OI-EE-2-CL-A and Figure 8.1-1, Revision 51
- Plant Electrical Distribution (125-Volt DC System), Checklist OI-EE-3-CL-A and Figure 8.1-1, Revision 51
- Auxiliary Feedwater, Checklist FW-4-CL-A and Drawing 11405-M-253, Sheet 1, Revision 76, and Sheet 4, Revision 1

Conclusions

The inspector found valves and switches to be in the correct position and power available to the valves, as appropriate. Some minor labeling deficiencies were noted and turned over to the licensee for action. None of the deficiencies noted had an impact on plant safety.

7. LER Followup (92700)

The following event reports were reviewed to verify that reportability requirements were fulfilled, corrective actions were accomplished, and actions were taken to prevent recurrence.

a. (Closed) LER 90-013: Alignment pin damage while moving the reactor head.

This LER was submitted by the licensee on a voluntary basis to report an incident of general interest. Toward the end of the 1990 refueling and maintenance outage, the licensee damaged the reactor vessel head flange and the vessel head alignment pins. During the placement of the head on the reactor vessel, the head was inadvertently lowered too far. It contacted the head alignment pins, bending the pins and causing damage to the head flange.

The root cause of the event was determined to be a deficiency in Procedure MM-RR-RC-0314, "Reactor Vessel Closure Head Installation," in that there was inadequate guidance for ensuring that an acceptable distance between the tops of the alignment pins and the bottom of the head was maintained. The procedure did require that the head, while over the vessel, not be more than 1 foot above the top of the pins. However, the vantage point provided for the signalman did not provide for a positive verification.

To prevent recurrence, the licensee committed to:

- Develop and implement a method of maintaining appropriate clearance between the reactor head and the tops of the alignment pins during head movement.

- Review Procedure MM-RR-RC-0314 to incorporate the above method and to assign an individual to watch for unexpected movement while the polar crane is stopped during the movement of the reactor head.

The inspector reviewed Revision 3 to MM-RR-RC-0314 and found the commitments to have been incorporated. The method of maintaining appropriate clearance is to station personnel more strategically and to fasten a rope with 1-foot graduations to the side of the head which will hang below the head. By observing the alignment pins and the rope graduations, personnel will be able to positively confirm the existence of required clearance.

b. (Closed) LER 90-017: Failure to perform local panel surveillance required by TS.

This LER was written due to the TS requiring local panel starting for the containment spray, safety injection, and shutdown cooling pumps. However, a remote operating panel for these pumps had never been installed, so no testing had been conducted in accordance with license requirements.

The licensee subsequently determined that there was no design requirement or licensing basis for a local panel. On December 3, 1990, the NRC issued Amendment 135 to remove the testing requirements from the TS.

c. (Closed) LER 90-020: Potential common-mode failure of the EDG exciter circuits.

This LER reported the potential for common-mode failure of the EDGs due to thermal failure or degradation of a voltage regulator in the static exciter circuitry. The condition had existed since original plant construction and was attributed to inadequate design of the static exciter control cabinets. Ventilation for the cabinets was not incorporated into the original designs. This condition resulted in internal cabinet temperatures detrimental to the voltage regulator.

This issue was part of an engineering analysis submitted to the NRC on September 12, 1990. The analysis and corrective actions taken and proposed by the licensee have been reviewed by the Region IV staff and found to be satisfactory.

The affected electronic components have been replaced, the cabinet has been ventilated via a temporary modification, and a permanent plant modification to ventilate the cabinets is scheduled for the next refueling outage.

Based on the actions taken by the licensee and the review performed by the Region IV staff, this LER is considered closed.

d. (Closed) LER 90-023: Safety injection piping and relief valves outside design basis.

This LER reported a condition in which the plant was discovered to be outside construction code requirements and thus outside the design basis of the plant. Specifically, the safety injection piping, bounded by the safety injection tank discharge isolation valves and the first check valve downstream of the isolation valves, did not conform to the design requirements of USAS-B31.7 in that the setpoint of Relief Valves SI-278, -279, -280, and -281 was found to be set at 395 psig. The piping they serve was designed to only 250 psig, with an initial hydrostatic test to 1.25 times the design value.

A violation was issued to address this issue. Since the corrective actions specified in the LER are the same as those specified in the licensee's response to Violation 285/9038-01, this LER is considered closed. The licensee's short-term corrective action was evaluated and documented in NRC Inspection Report 50-285/90-38. The licensee has committed to complete its long-term corrective action by March 31, 1991. Review of the licensee's long-term actions will be performed during followup of Violation 285/9038-01.

e. (Closed) LER 90-024: Failure to conduct an hourly firewatch.

This event concerned an hourly firewatch that was missed due to personnel error in transcribing the hourly firewatch log from one 24-hour period to the next. The error resulted in one fire door being listed twice and one fire door being left off the log. The fire door was left off the log for about 24 hours.

The licensee determined that the fire detectors were operable in the affected areas and that routine tours by plant personnel had occurred every 2 hours.

To prevent a recurrence of this event, the licensee implemented a computer-driven data base to generate the hourly firewatch log. This action appears adequate to resolve this concern.

f. (Closed) LER 90-027: Inadequate hourly firewatch patrols.

The licensee identified that firewatches were not being adequately performed because the entrance to a room with a degraded fire barrier had a radiological step-off pad installed in front of the door. The firewatch personnel were visually checking the door and the vent above the door for any signs of smoke or fire in lieu of entering the posted room. The licensee determined that this type of firewatch was inadequate. It was determined that the firewatch for some rooms had not been performed for approximately 1 month.

Attempts were made by the licensee to determine times when these areas or rooms were occupied, which would meet the intent of a firewatch. Taking into account the minimal available documentation, a rough estimate was that these areas were occupied at least hourly approximately 30 percent of the period when inadequate fire patrols were being performed. Fire detection instrumentation was determined to be operable throughout these periods.

The cause of this event was determined to be an inadequate understanding of the procedural requirements and inadequate direct supervisory guidance.

Corrective action included training in the method of proper firewatches and the proper method of inspecting doors within radiation controlled areas. These actions appear adequate to resolve this concern.

8. <u>Review of Previously Identified Inspection Findings (92701 and 92702)</u>

a. (Closed) Open Item 285/8922-08: Replacement of molded-case circuit breakers.

This item concerned the remaining action to be taken by the licensee in response to NRC Bulletin 88-10, "Nonconforming Molded-Case Circuit Breakers." The licensee had determined that four suspect breakers had been installed as output breakers in safety-related Inverters A, B, C, and D.

The licensee committed to replace these breakers during the 1991 refueling outage even though testing had found them to be adequate. However, the breakers were replaced during the recent outage caused by a leaking control element drive mechanism housing.

The inspector reviewed MWOs 893087, 893088, 893089, and 893090 for the breaker replacements. This adequately addressed this item.

b. (Closed) Violation 285/9013-03: Danger tag control problems.

This violation was issued for inadequate corrective action on the part of the licensee with respect to deficiencies found in its danger tag control problem. On March 20 and 21, 1990, the licensee's quality assurance organization identified problems with the control of danger tags, and plant management implemented corrective actions. The corrective actions were inadequate in that, on March 23, the inspector identified similar types of problems with the control of danger tags. In addition, on March 24 and 25, the licensee, during a followup review of the identified danger tag control problems, identified four additional instances where danger tags were not being controlled in accordance with licensee requirements.

In response to this violation, the licensee attributed the causes to inadequate training of maintenance and construction personnel prior to the 1990 refueling outage and inadequate verification of equipment tagging. The licensee provided additional training and revised Procedure SO-0-20, "Equipment Tagging," to require independent verification by a qualified operator of equipment tagging in preparation for equipment component outages for maintenance. At that time, July 1990, the licensee considered its corrective actions adequate.

Subsequently, on August 27, 1990, the licensee identified that another violation of the licensee's tagging procedure occurred despite the independent verification requirement. The licensee took aggressive short-term corrective actions and committed to study other licensees whose tagging programs have been identified as superior and to incorporate lessons learned by March 1, 1991. IFI 285/9038-03 was generated to track the licensee action.

Accordingly, Violation 285/9013-03 is considered closed and licensee action on its danger tag problem will be reviewed during routine followup of IFI 285/9038-03.

c. (Closed) Unresolved Item 285/9032-02: EDG upper temperature operating limits.

During the summers of 1989 and 1990, the licensee discovered that, during testing of the EDGs, the jacket cooling water reached an elevated temperature, causing an alarm. Further, it experienced a failure of a voltage regulator in the static exciter cabinet. The failure was attributed to elevated ambient temperatures.

Investigation and testing performed by the licensee determined that the upper limit for outside ambient air temperature was 107°F for EDG 1 and 103°F for EDG 2. Above these temperatures, the respective EDGs would not be able to carry their emergency loss-of-coolant accident loads and simultaneously maintain the 2000-hr Diesel Engine Manufacturers Association rating. The 2000-hr rating corresponds to loading which ensures that a high degree of reliability from the engine when operated at or below this rating.

With respect to the effect of elevated ambient temperatures on electrical components in the static exciter cabinet, the licensee incorporated a temporary modification to provide ventilation to the cabinets and initiated a permanent modification to modify the cabinets during the next refueling outage. See paragraph 7.c for a discussion on this issue.

Due to the complexities and magnitude of these issues, a formal engineering analysis was submitted to the NRC. The analysis was to specifically address an indepth and comprehensive evaluation of the EDG postaccident electrical loading and the effect of elevated temperatures on EDG operability. The analysis was submitted to the NRC on September 12, 1990.

The evaluation has been reviewed by the Region IV staff and no anomalies were noted. Based on the satisfactory results of the technical review, this item is considered closed.

d. (Closed) Unresolved Item 285/9038-02: Scaffolding tied off to a seismic support.

This item concerned the safety significance of unattended scaffolding that was found tied off to a safety-related seismic support. After the discovery by the inspector, the licensee untied the scaffold and issued Incident Report (IR) 900429.

In a memorandum dated January 25, 1990, the licensee's engineering department responded to the IR. The licensee concluded that no relevant forces could have been applied to the seismic support by the scaffold.

The inspector concluded that existing procedural controls for the use of scaffolding were adequate. This item appeared to be an isolated case since the inspector has not noted any similar occurrences.

9. Followup on TMI Items (25565)

a. (Closed) TMI Item I.A.1.3.1: Shift manning overtime limits.

The requirements of this item were published in Generic Letter 82-12. The licensee's implementation of these requirements are in TS 5.2.2.f and in Standing Order (SO) G-52, "Plant Staff Working Hours."

SO G-52 applies to the following plant personnel who perform safety-related functions: operations staff, shift technical advisors, shift health physics technicians, shift chemist, and key maintenance personnel (apprentice and above).

The requirements of SO G-52 specify that:

- An individual shall not be permitted to work more than 16 hours straight (excluding shift turnover time).

- An individual shall not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period (all excluding shift turnover time).

- A break of at least 8 hours shall be allowed between work periods (including shift turnover time). A work period is defined as 8 or more hours.

- Except during extended shutdown periods, the use of overtime shall be considered on an individual basis and not for the entire staff on shift.

Any deviations from the above can only be approved by the plant manager and must be documented on a Form FC-70.

Based on the inspector's review, it appeared that the licensee had adequately implemented the appropriate requirements to address this item.

b. (Closed) TMI Item I.C.6: Verify correct performance of operating activities.

This item requires that procedures ensure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human error and improving the quality of normal operations. This would reduce the frequency of the occurrence of situations that could result in or contribute to accidents. Such a verification system could include automatic system status monitoring, human verification of operations, and maintenance activities independent of the people performing the activity.

SO 0-20, "Equipment Tagging Procedure," incorporates the requirements of this TMI item. Specifically, SO 0-20 requires independent verification of lineups before and after tagging unless control room indication is available.

The inspector's review of SO 0-20 indicated that the requirements of this TMI item were satisfied. However, there was a recent tagging infraction documented in NRC Inspection Report 50-285/90-38. This indicated a weakness in the independent verification program. The licensee committed to review some plants with good tagging programs and revise the FCS program accordingly. This was scheduled to be completed by March 1, 1991. This item was made an inspector followup item in NRC Inspection Report 50-285/90-38.

This TMI item is considered closed based on the tagging program currently in place and the inspector followup of commitments made to improve the program.

c. (Closed) TMI Item II.B.1.3: Procedures for reactor coolant system (RCS) vents.

This item required the licensee to have implemented procedures for the venting of the RCS and reactor vessel. The purpose of the venting is to remove noncondensible gases from the RCS that may inhibit core cooling during natural circulation. The procedures are required to:

- Include information available to the operator for initiating or terminating vent usage.
- Define the conditions under which the vents should be used, as well as the conditions under which the vents should not be used.
- Ensure that the venting does not result in a violation of the requirements of 10 CFR Parts 50.44 or 50.46.
- Provide for removing noncondensible gases from the steam generators.

The inspectors reviewed the following licensee operating instructions (OI) to verify that applicable TMI venting requirements were proceduralized.

- OI-RC-2B, "Reactor Coolant Vent and Leak Test Instructions"
- OI-RC-2C, "Reactor Coolant System Cold Hydrostatic Test"
- OI-RC-2D, "Reactor Coolant System (RCS) Fill During Cold/Refueling Shutdown"
- OI-RC-3, "Reactor Coolant System (RCS) Startup"
- OI-RC-9, "Reactor Coolant Pump (RCP) Normal Operation"
- OI-CH-3, "Chemical and Volume Control System Normal Operation of the Volume Control Tank"

The above procedures adequately addressed the requirements of this TMI item.

10. Exit Interview

The inspectors met with Mr. W. G. Gates (Division Manager, Nuclear Operations) and other members of the licensee staff on February 26, 1991. 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.