# APPENDIX

## U.S. NUCLEAR REGULATORY COMMISSION
## REGION IV

NRC Inspection Report: 50-285/90-45 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: December 5, 1990, through January 15, 1991

Inspectors: R. Mullikin, Senior Resident Inspector
T. Reis, Resident Inspector
R. Azua, Project Engineer

Approved: P. H. Harrell, Chief, Project Section C 1-24-91
Date

Inspection Summary

Inspection Conducted December 5, 1990, through January 15, 1991 (Report 50-285/90-45)

Areas Inspected: Routine, unannounced inspection of onsite followup of events, operational safety verification, surveillance and maintenance observations, licensee event report followup, review of previously identified inspection findings, and midcycle plant performance review.

Results:

- The licensee's actions regarding the decision to shut down the plant and locate the source of unidentified reactor coolant system leakage was conservative. Throughout the shutdown, the licensee exhibited an obvious safety consciousness. In addition, communication with the inspectors was very good (paragraph 3.a).

- The licensee appeared to not be familiar with requirements for the performance of leak rate testing of a containment isolation valve (paragraph 3.b).

- Adequate implementation of the radiation protection and security programs was noted (paragraphs 4.c and 4.d).

- Implementation of the maintenance and surveillance programs appeared to be adequate (paragraphs 5 and 6).

- A midcycle performance review was held with the licensee (paragraph 9). The review outline is provided as an attachment to this inspection report.

DETAILS

1. Persons Contacted

Principal Licensee Employees

#*R. L. Andrews, Division Manager, Nuclear Services
*C. J. Brunnert, Supervisor, Operations Quality Assurance
#*J. W. Chase, Manager, Nuclear Licensing and Industry Affairs
#F. F. Franco, Manager, Radiological Services
#*S. K. Gambhir, Division Manager, Production Engineering
#*J. K. Gasper, Manager, Training
#*W. G. Gates, Division Manager, Nuclear Operations
*L. T. Kusek, Manager, Nuclear Safety Review Group
*D. J. Matthews, Supervisor, Station Licensing
#*T. L. Patterson, Manager, Fort Calhoun Station
#*H. J. Sefick, Manager, Security Services
*R. W. Short, Supervisor, Special Services Engineering
*C. F. Simmons, Station Licensing Engineer
#*S. J. Willrett, Manager, Nuclear Materials/Administration

NRC Personnel

#T. P. Gwynn, Acting Director, Project Directorate IV-1, Office of Nuclear Reactor Regulation
#P. H. Harrell, Chief, Project Section C

The inspectors also contacted additional personnel during this inspection period.

#Denotes attendance at the midcycle plant performance review meeting conducted on January 15, 1991 (see paragraph 9).

*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 Decer.er 14, 1990, when the licensee commenced a controlled reduction in power. The power reduction was initiated to perform an inspection of the reactor vessel head area to identify the source of unidentified reactor coolant system leakage.

The licensee discovered, during the inspection, that a spare control element drive mechanism (CEDM) housing had a through-wall crack. This and the other similar spare CEDM housing were removed and inspected. Both housings were replaced with blank flanges.

On January 2, 1991, the licensee commenced a plant startup, but a leaking flanged connection near a primary code safety valve (RC-142) required that the plant be returned to cold shutdown. The leak was corrected, except for a small amount of weeping, and the startup was resumed.

The plant achieved criticality on January 8; full power was achieved on January 14 and maintained through the end of this inspection period.

3. Onsite Followup of Events (93702)

a. Through-Wall Leak in CEDM 9

On December 14, 1990, at approximately 7:30 p.m. (CST), the licensee commenced a reduction in power from 100 percent to locate the source of unidentified reactor coolant system (RCS) leakage. The plant entered hot standby (Mode 2) on December 15 at 4:26 a.m.

The FCS experienced increasing unidentified RCS leakage in the range of 0.15 to 0.40 gpm since October 21, 1990, which was below the Technical Specification (TS) allowable limit of 1.0 gpm. Initial attempts at locating the source of the leakage were unsuccessful.

The licensee formed, in November 1990, a small internal group to attempt to locate the source of the RCS leakage. The group performed walkdowns inside and outside of containment and found no external leakage, no unexpectedly warm leakoff lines, and no other indications of abnormal leakage. The group determined that the waste holdup (WHU) tanks, spent regenerant tanks (SRT), and the containment sump showed some evidence of reactor coolant. However, little activity was identified in the containment sump. The preliminary conclusion was that an uncollected reactor coolant leak inside containment existed. Also, the group concluded that, due to the leak, humidified air was being cooled by the containment cooling units and the condensed moisture was draining to the containment sump. In this distillation process, nearly all activity, except tritium, was removed from the condensed water along with boron. The condensed water collected in the sump was pumped to the SRT and then pumped to the WHU tanks. In addition, containment air samples showed radioisotopes not normally present.

Based upon the results of the investigation and intermittent alarms received on fire detectors near the cooling fans for the CEDMs, the licensee determined that a primary leak probably existed in the area of the reactor vessel head. Thus, licensee management decided, on December 14, to reduce power and perform an inspection of the reactor vessel head area. On December 15 a containment entry was made and boric acid crystals were discovered around CEDM 9. A leaking weld was initially suspected as the source of the RCS leakage. Due to this finding, the licensee decided to continue the plant cooldown. On the same day, a second containment entry was performed to remove enough of the boric acid crystals to pinpoint the source of the leak. Removal of the crystals indicated that a weld was not leaking, but a through-wall defect in the CEDM 9 housing probably existed.

CEDM 9 is one of four spare CEDM housings that were designed for additional reactivity control in the event that plutonium recycle fuel was to be used. Two of the spare housings (CEDMs 9 and 13) contain a natural circulation spoiler that prevents hot reactor coolant from reaching the top of the housing. The other two spare housings (CEDMs 7 and 11) provide a means for the heated junction thermocouples (HJTC) to enter the reactor vessel for the vessel level monitoring system. The licensee initially suspected that the spoiler may have eroded the housing, and plans were made to use a borescope to inspect the inside of the CEDM 9 housing.

Combustion Engineering (CE) representatives arrived on site on December 18, to perform the inspection of CEDM 9. During the night shift on December 18, the licensee removed the cap from the CEDM housing and was able to remove the flow spoiler from the housing. An initial inspection indicated that the spoiler was not the cause. On December 19, the licensee completed the removal of the CEDM/vessel head seal weld, and the CEDM 9 housing was successfully removed for CE to perform a borescopic inspection of the housing internals.

The results of the borescopic inspection showed a 3/4-inch axial crack on the outside of the CEDM 9 housing and a 2 7/8-inch crack on the inside. In addition, another indication was found approximately 120 degrees from the through-wall crack. This indication was on the inside and was about 2 inches long. The through-wall penetration was near the weld overlay buildup on the inside surface of the housing. This inspection could not identify the cause of the cracking.

The licensee decided to remove the other spare housing (CEDM 13) with the flow spoiler. This was accomplished without incident and the borescopic inspection, performed on December 21, revealed three axial cracks in the area of the weld overlay, similar to those observed on CEDM 9. Sample sections of CEDMs 9 and 13 were cut from the housings and shipped to CE in Windsor, Connecticut, for a more detailed analysis.

Based upon CE's detailed inspection, which included visual and ultrasonic (UT) inspections, only two cracks were confirmed in each housing. Additional linear indications reported from inspections at the site were revealed to be surface scratches or machining marks that did not penetrate into the housing. An additional analysis revealed that one through-wall and one 86 percent through-wall crack existed in CEDM 9. The two cracks in CEDM 13 were 98 percent and 70 percent through-wall.

Fractographic examination of the crack surfaces revealed that all of the cracks had initiated on the inside diameter surface, sometimes with more than one initiation site. The initiation sites were all near the upper edge of the weld overlay region. The cracks then propagated outward into the wall of the CEDM housing, extending nearly symmetrically downward through the weld overlay region and upwards into the base metal of the housing. The welds were oriented in a nominally axial direction, but some of the cracks and portions of cracks were skewed axially by about 15 degrees.

Scanning electron microscopy of the crack surfaces and metallographic analysis of cross sections of the crack were performed to identify the mode of cracking. These evaluations revealed the cracking to be transgranular stress corrosion cracking (TGSCC).

Based upon the metallurgical evaluation of CEDMs 9 and 13, the cause of the TGSCC was concluded by the licensee and CE to be the presence of locally high oxygen concentrations in the unvented CEDM housings. It was discovered that the high oxygen content, combined with low chloride levels, and the combined residual and pressure induced tensile hoop stresses were sufficient to cause TGSCC in the solution-annealed, Type-348 stainless steel pressure housing. The licensee determined that CEDM 9 and 13 housings have not been vented since initial startup of the plant in 1973.

The licensee determined that CEDMs 7 and 11, with the HJTC probes installed, had been fully vented since 1984, when the probes were installed. It was determined that the venting procedures used by the licensee may not have ensured that these housings were free of air bubbles. Based on this information, the licensee decided to examine CEDMs 7 and 11 by UT to detect the presence of any cracks. The licensee contracted with EBASCO Services, Incorporated to perform the UT inspection of CEDMs 7 and 11. This inspection was performed on December 29, and no indications of cracks were discovered.

The 37 active (i.e., containing a drive mechanism) CEDM housings have a mechanical seal installed that allows air to vent during RCS fill and during CEDM operation. Based on the finding that no indications were found in CEDMs 7 and 11, the licensee opted not to perform a UT inspection of the 37 active CEDMs since these CEDMs are continuously vented.

The licensee inspected the reactor vessel for boric acid damage to ensure that no boric acid corrosion of carbon steel components had occurred. The inspection identified no degradation of the carbon steel materials.

Following the decision to remove CEDMs 9 and 13, CE designed and fabricated CEDM blank flanges to replace the removed housings. These blank flanges have no venting capability, but were deemed acceptable by CE based on two reasons. The first was that the blank flanges do not have a weld overlay region, therefore, no residual stresses to assist in the initiation of stress corrosion cracking. Secondly, without the natural circulation spoiler, there is greater expectation that the oxygen level in the shorter configuration will be closer to the bulk coolant chemistry levels as a result of more natural circulation.

The flanges were installed in accordance with Modification Request (MR) FC-90-74, "CEDM 9 and 13 Modification." The inspector reviewed the design package and the 10 CFR Part 50.59 evaluation prepared by the licensee. The inspector noted that the licensee had demonstrated that the permanent plant modification did not constitute an unreviewed safety question. Further, the modification design was reviewed by a Region IV materials and quality assurance engineering specialist and no problems were noted.

On January 2, 1991, the licensee commenced a plant startup. Before reaching criticality, a hot hydrostatic test was performed to verify that the CEDM 9 and 13 blank flanges were not leaking. This test showed no leaking from the flanges, but a leak was detected, by visual observation, from a flanged joint near a primary code safety valve (RC-142). A closer inspection revealed that there was a leaking flange gasket on the the pressurizer side of the loop seal. Initial attempts at stopping the leak were unsuccessful and the licensee decided to return to cold shutdown and replace the flange gasket.

On January 4 the flange was disassembled and found to be damaged. It was determined that, during previous assembly of the flange, the tongue and groove assembly on the flange surfaces did not mate correctly. Thus, when the flange was reassembled and torqued, the tongue and groove were damaged. This misalignment had not been noticed during assembly and the flange was successfully leak tested upon plant startup. The tongue and groove were repaired and the flange reassembled. However, the flange was leak tested and found to be leaking slightly. The licensee decided to resume the plant startup and monitor the small leak during weekly containment entries.

The licensee revised Procedure OP-ST-RC-3001, "Reactor Coolant System (RCS) Leak Rate Test," to require notification of plant management if unidentified RCS leakage is 0.20 gpm or greater for 3 consecutive days. If the leakage reaches this criteria, management will decide whether to shut down the plant or continue operating. The highest leak rate experienced, by the end of this inspection period, was 0.08 gpm.

The inspectors will review further licensee actions during routine followup of Licensee Event Report (LER) 90-028, which will be issued by the licensee to document the details of this issue.

b. Repair of a Containment Isolation Valve

On December 4, 1990, RCS sample line Valve HCV-2504B, a containment isolation valve, was removed for repairs due to a packing leak and a body-to-bonnet gasket leak.

The inspector verified that the valve was properly isolated and tagged to ensure that containment integrity was maintained. When the component was physically removed from the system for repair, the licensee lost the capability to sample the RCS for gross radioactivity.

TS 3.2 requires that the RCS be sampled for gross radioactivity, while in Mode 1, once every 3 days. A sample was obtained just prior to removing Valve HCV-2504B from service on December 4. This required the licensee to complete an RCS sample by 3 a.m. on December 8, which includes a TS allowance of plus 25 percent.

The inspector followed the licensee's repair efforts for Valve HCV-2504B and reviewed Maintenance Work Order (MWO) 904627, "Packing Leak - HCV-2504B," and Maintenance Procedure PE-RR-VX-0426S, "Inspection and Repair of Safety Related Masoneilan Air-to-Open Model 38-20571 Nuclear Valves." Weaknesses were noted in the licensee's management of the repair of the valve. No "Req'd Inservice Date" or "LCO Action" was designated in the work release section of MWO 904627. Also, the licensee indicated that it was doubtful that Surveillance Test IC-ST-CONT-1001, "Containment Isolation Valves Leakage Rate Test, Type C," could be successfully performed due to seat leakage past upstream Valve HCV-2504A. The licensee intended to make a one-time procedure change to allow the local leak rate test to be performed with water in lieu of the procedurally required nitrogen. The licensee was apparently unaware that TS 3.5(4)(b) specifically requires that Type C tests be performed with air or nitrogen. Ultimately, the valve was repaired and reinstalled, and satisfactory postmaintenance testing was performed within the TS time limit.

The deficiencies noted above, with the completion of the MWO and the TS requirements for local leak rate testing, appeared to be isolated cases and not indicative of programmatic problems. The inspectors will continue to review this area during the performance of routine inspection activities in the future.

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 the they were aware of plant status and the reasons for lit annunciators. A review of the shift turnover sheets indicated that TS limiting conditions for operation correctly reflected equipment status. No problems were noted during control room observations.

b. Plant Tours

The inspectors routinely toured various areas of the plant to verify that proper housekeeping was being maintained. The inspectors noted that the licensee's plant upgrade project had focused on the radiologically controlled area (RCA). The RCA was in the process of being painted throughout this inspection period.

c. Radiological Protection Program Observations

During this inspection period, the inspectors witnessed that proper radiation protection practices were being maintained. The inspectors verified that personnel were observing proper RCA entrance and exit requirements. In addition, health physics (HP) personnel and a HP supervisor were observed in the RCA, on a routine basis, to ensure that the radiation protection program was being properly implemented.

d. Security Program Observations

The inspectors verified that selected activities of the licensee's security program were being adequately implemented. Personnel access requirements were found to be adequately performed. During tours of the plant, the inspectors noted that compensatory measures were implemented when security doors were required to be open for maintenance activities.

On December 20, 1990, the inspector observed a security guard performing her watch duties of two plant personnel on the veranda of the intake structure. The workers were attempting to prevent the clogging of the intake screens by ice on the Missouri River. The dedication of these individuals was noteworthy due to the extreme cold conditions encountered by these licensee personnel. The actual temperature at the time was approximately -10°F, with a wind chill factor of about -40°F.

e. Monitoring of the Missouri River Level

During this inspection period, the Missouri River level was maintained at a fairly steady state. The Army Corps of Engineers maintained a release rate of 10,000 cubic feet per second from the Gavins Point Dam, which is upstream of the FCS. The TS requires that, when river level reaches 980 feet (mean sea level), a continuous monitoring of river level be performed, and at a level of 976 feet 9 inches, a plant shutdown is to be commenced. During this period, the river level dropped to approximately 982 feet. It appeared that, barring a sudden drop in river level due to ice jams upstream, the Missouri River level should not adversely effect plant operations during this winter. Plant personnel were found to be very cognizant of the status of the river level.

5. Maintenance Observations (62703)

a. Raw Water Pump Strainer Replacement

The inspector observed, on January 3 and 4, 1991, selected portions of maintenance activities associated with the replacement of raw water pump Strainer AC-12B. The work was performed under MWO 902240 using Maintenance Procedure MM-RR-RW-0200, "Removal and Installation of Raw Water Strainers." The inspector reviewed the work in progress, the completed installation, and the completed documentation. It was noted that the required flame and welding permit was complete and visible at the site of the work.

Review of the completed documentation showed that no quality control hold points were missed, and postmaintenance testing was successfully completed.

b. Maintenance Team Reinspection

During the period of December 1-20, 1990, a team of inspectors from the Region IV office performed a reinspection of maintenance activities at the FCS. The results of this inspection will be documented in NRC Inspection Report 50-285/90-36.

6. Surveillance Observations (61726)

On January 8, 1991, the inspector witnessed the performance of Procedures OP-ST-ESP-0009, "Channel A Safety Injection, Containment Spray and Recirculation Actuation Signal Test"; OP-ST-CEA-0001, "Control Element Assembly Position Indicating System (CEAPIS) Check"; and IC-ST-AFW-0001, "Auto Initiation of Auxiliary Feedwater Functional Check of Initiation Circuits."

Procedure OP-ST-ESF-0009 was performed to satisfy the requirements of TS 3.1 for the applicable safeguards actuation signals.
Procedure OP-ST-CEA-0001 was performed to ensure that the secondary CEAPIS was accurate by direct comparison with the primary CEAPIS.
Procedure IC-ST-AFW-0001 verified operability of the automatic initiation of auxiliary feedwater circuits.

All three of the surveillances were accomplished by qualified personnel, in accordance with the approved test procedures. The test data was found to be accurate and complete, and the test results met the TS requirements. Following completion of the tests, the inspector independently verified that all the systems effected by the surveillance tests were properly returned to service.

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 90-002: Inadvertent actuation of the containment isolation actuation signal (CIAS).

On February 26, 1990, while the plant was in a refueling shutdown condition, a CIAS was generated following an accidental actuation of a lockout relay by a craftsman working in a control room cabinet. The cause of this event was the failure of the design and modification process to anticipate the difficulty that the craftsman would encounter while performing the modification.

As a result of this event, the licensee issued a memorandum to the appropriate plant personnel reminding them to use caution and proper work practices when working around electrical control equipment. In addition, a review of engineering modification procedures was performed to ensure that guidance regarding potential inadvertent actuation of circuitry was appropriately addressed. Finally, plant management evaluated the present administrative guidance and controls for work that could effect electrical control equipment.

b. (Closed) LER 90-006: Loss of offsite power and diesel generator actuation.

The corrective actions taken for LER 90-002 (above) also implement the actions required to address this issue; therefore, this LER is considered closed.

c. (Closed) LER 90-008: Inadvertent actuation of pressurizer pressure low signal.

The corrective actions taken for LER 90-002 (above) also implement the actions required to address this issue; therefore, this LER is considered closed.

8. Review of Previously Identified Inspection Findings (92701 and 92702)

a. (Closed) Deviation 285/9002-03: Failure to properly install electrical cables in trays.

This deviation concerned instrumentation and control (I&C) cables installed in safety-related Trays 5-4A and 5-4B that did not comply with the requirements provided in Figure 8.5-1 of the Updated Safety Analysis Report (USAR). The USAR stated that cable tray fill for 125-Vdc and 120-Vac cables shall generally not exceed a maximum of 50 percent and that safety- and nonsafety-related cables, in the same tray, shall be separated by metallic barriers. In addition, I&C cables are required to be tied down in a neat configuration after installation in the trays.

The inspector found that the volume of cables in Trays 5-4A and 5-4B was so great that the cables extended above the metallic barriers separating the raceways in the cable trays. Also, the vertically run cables in these trays were not tied down in a neat configuration.

The licensee's corrective actions included:

- Using existing records to determine which trays may have an overfill situation.

- Performing an inspection of accessible safety-related cable tray sections where overfill was identified.

- Reworking of accessible areas of Trays 5-4A and 5-4B to neatly tie down cables and remove crossovers of the metallic barrier.

- Preparing an engineering analysis to justify the existing cable tray configuration (fill and power cable derating).

- Updating of Engineering Instruction GEI-9, "Electrical System Interactions," to require specific analysis for each modification that involves the installation or change in routing of cables.

- Updating of Construction Procedure ETS-10, "Cable Installation Specification," to provide better instructions to the craftsmen and quality control inspectors on cable installation.

The inspector reviewed Engineering Analysis (EA) FC-90-76 and the revisions to Procedures GEI-9 and ETS-10. The licensee stated that EA FC-90-76 will be used as the technical basis for a future revision to the USAR that will address acceptable cable tray loading. The analysis appeared comprehensive as it considered the effects of cable heating, physical separation, and electromagnetic interference on cable tray fill for I&C, control, and power cables. Even though not all of the overfilled trays were accessible, the licensee was able to determine that a safety concern did not exist for these trays even if crossover of cables existed.

In addition, the inspector toured various areas of the plant to look for other overfill conditions that may not have been recognized by the licensee. The inspector could find no other examples of overfill conditions.

b. (Closed) Open Item 285/8934-01: Scheduling of Generic Letter 88-17 related instrument modifications and the sequencing of implementation of the appropriate revised procedures.

The licensee's response to Generic Letter 88-17 failed to address the schedule for installing level indication instrumentation, and it did not identify when procedure modifications would be implemented.

The following maintenance procedures were changed to provide appropriate instructions for the installation and removal of the temporary core exit thermocouple cables:

- MP-RC-10-8-A, "Procedure for the Removal of CEDM Cables, Trays, HJTC and Incore Detector Cables, CEDM Cooling Ducts, and CEDM Cooling Pipe," dated February 22, 1990
- MP-RC-10-8-B, "Procedure for the Replacement of CEDM Cables, Trays, HJTC and Incore Detector Cables, CEDM Cooling Ducts, and CEDM Cooling Pipe," dated April 20, 1990
- MP-RC-6-4-A, "Removal of ICI Bullet Nose Assemblies," dated April 20, 1990
- MP-RC-6-4-B, "Installation of ICI Bullet Nose Assemblies," dated March 16, 1990

Modification Package FC-89-25, dated March 1, 1990, directed the installation of the RCS level indication instrumentation. This installation was completed on April 12, 1990. In addition, the modification package addressed the changes required to be implemented in the operating instructions and Abnormal Operating Procedure AOP-19, "Loss of Shutdown Cooling." These procedure changes were implemented as part of the modifications.

9. Midcycle Plant Performance Review (35502)

On January 15, 1991, a conference was held on site between the licensee and Region IV personnel to discuss the NRC's evaluation of the licensee's performance at the approximate midpoint of the current SALP cycle. The licensee's current SALP cycle extends from May 1, 1990, to July 31, 1991.

The NRC conducts a performance review to provide feedback to the licensee on the current status of their performance. The attendees at the meeting are listed in paragraph 1. The outline of the information presented at the meeting is included as an attachment to this inspection report.

10. Exit Interview

The inspectors met with Mr. W. G. Gates (Division Manager, Nuclear Operations) and other members of the licensee staff on January 15, 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.

ATTACHMENT

MIDCYCLE PERFORMANCE REVIEW

FORT CALHOUN STATION

JANUARY 15, 1991

PLANT OPERATIONS

- ONSHIFT OPERATIONS STAFF RESPONSE TO MINOR PLANT EVENTS AND PERTURBATIONS WAS VERY GOOD

  - Rupture of instrument air header

  - Degradation of a reactor coolant pump seal - Concern identified with the retention of physical evidence

  - Responses were efficient, correct, and conservative

  - Indication of high experience and knowledge level

- MANAGEMENT'S CONSERVATIVE APPROACH TO PLANT SAFETY WAS APPARENT

  - Plant shut down when raw water, containment spray, and component cooling water were found to be outside design basis until issue could be resolved

  - Through-wall leak in a control rod drive mechanism housing

  - Apparent conservative approach toward equipment operability and event reportability issues

  - Upgrade of the schedule for the 10-year inservice inspection due to concerns with the thermal shield

- HOUSEKEEPING AND PLANT APPEARANCE CONTINUED TO BE VERY GOOD

- MANAGEMENT SUPPORT OF OPERATIONS NEEDS ADDITIONAL ATTENTION

  - Onshift communications have been poor and communication standards have not been established. Training teaches but is not implemented on shift.

  - Control of emergency operating procedures done by training instead of operations, atypical of other Region IV plants.

  - Lack of guidelines for the plant conditions that require entry into emergency operating procedures.

- ONSHIFT OPERATIONS STAFF PERFORMANCE OF EMERGENCY OPERATING PROCEDURE ACTIONS, WITH THE MINIMUM TECHNICAL SPECIFICATION REQUIRED STAFFING LEVELS, WAS AVERAGE

- LEVEL OF ONSHIFT STAFFING HAS BEEN IMPROVED

- MANAGEMENT TOURS OF THE PLANT, OTHER THAN THE CONTROL ROOM, HAVE BEEN WEAK

- OVERALL

  - Onshift operations crew performance very good for minor plant perturbations and events.

  - Management approach to safe plant operation has been conservative

  - Operations management should become proactive in setting and enforcing standards, such as onshift communications, and in resolving emergency operating procedures issues related to the performance of onshift operators.

RADIOLOGICAL PROTECTION

- IMPLEMENTATION OF RADIOLOGICAL PROTECTION PROGRAM WAS VERY GOOD

  - Minor personnel errors

    - Inadequate survey of fuel transfer tube

    - Entry into high radiation area without proper dosimetry

  - Errors appear to be isolated and not indicative of programmatic problems

- POSTOUTAGE CLEANUP EFFORTS WERE VERY GOOD

- OVERALL

  - Efforts in this area continue to be very good

MAINTENANCE/SURVEILLANCE

- ADEQUATE IMPLEMENTATION OF THE MAINTENANCE PROGRAM WAS NOTED

- CONCERNS WERE IDENTIFIED IN THE POSTMAINTENANCE TESTING AREA

  - Minor problem noted when diaphragm replaced in air regulator by a preventive maintenance activity and no postmaintenance testing was done

  - Adequacy of postmaintenance testing instructions in preventive maintenance activities weak

- Other than this problem, appears previous problems identified with postmaintenance testing have been resolved

° MAINTENANCE TEAM REINSPECTION RESULTS WERE FAVORABLE

- No major issues identified

- Concerns were noted with the verification and validation process used for maintenance procedures

- Some programs in this area are in the process of being implemented and will be reviewed in the future

° SURVEILLANCE PROGRAM IMPLEMENTATION WAS VERY GOOD

- Minor problem noted with failure to verify emergency diesel generator loads did not exceed 2000 hr-kW rating after 1990 outage

- The problem was initially identified in the 1988 outage and licensee failed to adequately correct it

- Missed surveillances have not been a problem as was the case in the past

- Effectively implementing program

° MATERIAL CONDITION OF PLANT VERY GOOD

° IMPROVEMENT IN MANAGEMENT OVERSIGHT OF MAINTENANCE NOTED

° OVERALL

- Additional management attention may be needed for postmaintenance testing requirements in preventive maintenance activities

- Timely and proactive resolution of the items identified by the maintenance team should be accomplished

- Review of the validation and verification process for maintenance and other procedures, as necessary, should be initiated

- Management oversight of maintenance activities has been notably improved

- Material condition of the plant has been very good

EMERGENCY PLANNING

° ANNUAL EXERCISE IDENTIFIED CONTINUING PROGRAM WEAKNESSES

- Poor response by the technical support center staff

- Untimely response to fire
- Inadequate personnel access control to the protected area
- Poor information flow from the control room
- Problems with the scenario

° OVERALL

- Problems continue to be identified with implementation of the emergency preparedness program
- Management should resolve these problems in a timely and effective manner

SECURITY

° PROGRAM IMPLEMENTATION HAS BEEN ADEQUATE

- Fitness for duty inspection identified no problems

° OVERALL

- No problems noted

ENGINEERING/TECHNICAL SUPPORT

° DESIGN BASIS RECONSTITUTION PROGRAM WAS A CONTINUING STRENGTH

- Program continues to identify and correct problems due to proactive efforts by system and design engineering personnel
- Program one of the best in Region IV
- Minor problems noted that should have been identified
  . Safety injection tank relief valve set too high
  . Seismic qualification of auxiliary steam system in the emergency diesel generator rooms

° FIRE PROTECTION/PREVENTION PROGRAM IMPLEMENTATION ADEQUATE

- Appears past problems with implementing fire watch patrols have been adequately addressed

° EMERGENCY OPERATING PROCEDURE TEAM INSPECTION IDENTIFIED CONCERNS

- Emergency operating procedures were adequate

- Validation and verification process for emergency operating procedures was weak

- Basis of the development of site specific emergency operating procedures from generic Combustion Engineering guidance was inadequate

° OVERALL

- Strong reconstitution program

- Strong engineering program

- Management should resolve emergency operating procedure inadequacies in a timely manner

SAFETY ASSESSMENT/QUALITY VERIFICATION

° SAFETY ENHANCEMENT PROGRAM IMPLEMENTATION INDICATES POSITIVE RESULTS

- Positive results with oversight and management of plant have been noted with the implementation of the completed items

- After receipt of January 1991 letter from OPPD with final Safety Enhancement Program update, region will develop closeout plan

° QUALITY OF TECHNICAL SPECIFICATION AMENDMENT SUBMITTALS HAS IMPROVED

° QUALITY ASSURANCE IS IDENTIFYING PROBLEMS

- Lubricating oil not being purchased as safety-related material

° OVERALL

- Continuing good performance

NRC FORM 760
IE MC 0535

U.S. NUCLEAR REGULATORY COMMISSION

# INSPECTOR'S REPORT
Office of Inspection and Enforcement

PRINCIPAL INSPECTOR (Name, last, first, and middle initial): MULLIKIN, R. P.
REVIEWER: HARRELL, P. H.

INSPECTORS: REIS, T.    AZUA, R.

| LICENSEE/VENDOR | TRANSACTION TYPE | DOCKET NO. (8 digits) OR LICENSE NO. (BY PRODUCT) (13 digits) | REPORT NO. | SEQ | NEXT INSPEC. DATE MO | YR |
|---|---|---|---|---|---|---|
| OMAHA PUBLIC POWER DISTRICT | X I – INSERT; M – MODIFY; D – DELETE; R – REPLACE | 05000285 | 9045 | A B C D | | |

| PERIOD OF INVESTIGATION/INSPECTION FROM (MO DAY YR) | TO (MO DAY YR) | INSPECTION PERFORMED BY | ORGANIZATION CODE OF REGION/HQ CONDUCTING ACTIVITY (See IEMC 0530 Manpower Reporting—Weekly Manpower Reporting for code.) |
|---|---|---|---|
| 12 05 90 | 01 15 91 | 1 – REGIONAL OFFICE STAFF; X 2 – RESIDENT INSPECTOR; 3 – PERFORMANCE APPRAISAL TEAM; OTHER | REGION 4, DIVISION 6, BRANCH 2 |

REGIONAL ACTION (Check one box only): 1 – NRC FORM 591; X 2 – REGIONAL OFFICE LETTER

TYPE OF ACTIVITY CONDUCTED (Check one box only): X 02 – SAFETY; 03 – INCIDENT; 04 – ENFORCEMENT; 05 – MGMT. AUDIT; 06 – MGMT. VISIT; 07 – SPECIAL; 08 – VENDOR; 09 – MAT. ACCT.; 10 – PLANT SEC; 11 – INVENT. VER.; 12 – SHIPMENT/EXPORT; 13 – IMPORT; 14 – INQUIRY; 15 – INVESTIGATION

INSPECTION/INVESTIGATION FINDINGS (Check one box only): A: X 1 – CLEAR; 2 – VIOLATION; 3 – DEVIATION; 4 – VIOLATION & DEVIATION

TOTAL NUMBER OF VIOLATIONS AND DEVIATIONS: A 00

ENFORCEMENT CONFERENCE HELD: 1 – YES

REPORT CONTAIN 2.790 INFORMATION: 1 – YES

LETTER OR REPORT TRANSMITTAL DATE: NRC FORM 591 OR REG. LETTER ISSUED: JAN 28 1991; REPORT SENT TO HQ. FOR ACTION: MO DAY YR

MODULE INFORMATION

| REC. ORD. | MODULE NUMBER INSP. | DIRECT INSPEC. EFFORT (STAFF HOURS) | PERCENTAGE COMPLETED TO DATE | STATUS | MODULE REQ. FOLLOWUP | DESCRIPTION |
|---|---|---|---|---|---|---|
| 01 | 3 71707 | 017 | 100 | C | | OPS. SAFETY VERIF. |
| 02 | 5 62703 | 004 | 100 | C | | MONTHLY MAINT. |
| 03 | 5 61726 | 017 | 100 | C | | MONTHLY SURV. |
| 04 | 5 92700 | 017 | 100 | C | | LER FOLLOWUP |
| 05 | 5 92701 | 004 | 100 | C | | FOLLOWUP |
| 06 | 5 92702 | 008 | 100 | C | 5 71707 | FOLLOWUP VIOL & DEV |
| 07 | 5 93702 | 039 | 100 | C | | ONSITE EVENTS |
| 08 | 5 35502 | 002 | 100 | C | | MIDCYCLE REVIEW |

* DIRECT SEQUENCE IF VIOLATION OR DEVIATION