APPENDIX

U. S. NUCLEAR REGULATORY COMMISSION

REGION IV

NRC Inspection Report: 50-285/84-13 License: DPR-40

Docket: 50-285

Licensee: Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Facility Name: Fort Calhoun Station

Inspection At: Fort Calhoun Station, Blair, Nebraska

Inspection Conducted: May 1-31, 1984

Inspector: L A Yandell 6/19/84
L. A. Yandell, Senior Resident Reactor Inspector Date

Inspector: J P Jaudon 6/27/84
for D. P. Tomlinson, Reactor Inspector Date

Approved: J P Jaudon 6/27/84
J. P. Jaudon, Acting Chief, Project Section A, RPB2 Date

Inspection Summary

Inspection Conducted May 1-31, 1984 (50-285/84-13)

Areas Inspected: Routine, announced inspection of operational safety verification, surveillance testing, maintenance activities, and followup of steam generator tube failure incident. The inspection involved 178 inspector-hours onsite by two NRC inspectors.

Results: Within the four areas inspected, no violations or deviations were identified.

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DETAILS

1. Persons Contacted

   *W. G. Gates, Manager, Fort Calhoun Station
   M. R. Core, Supervisor, Maintenance
   L. T. Kusek, Supervisor, Operations
   A. W. Richard, Supervisor, Technical
   J. J. Fisicaro, Licensing Administrator Supervisor, Nuclear Regulatory and Industry Affairs
   T. J. McIvor, Manager, Operations-Technical Support Services
   J. K. Gasper, Manager, Reactor & Computer Technical Services
   R. L. Jaworski, Section Manager, Technical Services
   W. C. Jones, Division Manager, Production Operations
   K. J. Morris, Manager, Adminstrative Services
   C. W. Norris, Senior Engineer, Nuclear & Chemical Technical Services

   *Denotes attendance at the exit interview.

   The NRC inspectors also talked with and interviewed, other licensee employees during the inspection. These employees included licensed and unlicensed operators, craftsmen, engineers, and office personnel.

2. Operational Safety Verification

   The NRC inspector performed activities as described below to ascertain that the facility is being maintained in conformance with regulatory requirements and that the licensee's management control system is effectively discharging its responsibilities for continued safe shutdown.

   a. The NRC inspector made several control room observations to verify proper shift manning, operator adherence to approved procedures, and adherence to selected Technical Specifications specific to the shutdown condition. Selected logs, records, recorder traces, annunciators, panel indications, and switch positions were reviewed to verify compliance with regulatory requirements. Radiation controlled area access points were observed at various times to verify that they were being maintained in accordance with approved procedures. The licensee's equipment control was reviewed for proper implementation by reviewing the maintenance order and tag-out logs, and by verifying selected safety-related tag-outs. The NRC inspector observed several shift turnovers and attended a number of the outage planning meetings.

   b. The NRC inspector toured the plant at various times to assess plant and equipment conditions. The following items were observed during these tours:

      . general plant conditions

- vital area barriers not degraded or appropriately manned by security personnel
- adherence to requirements of radiation work permits (RWPs)
- proper use of protective clothing and respirators
- plant housekeeping and cleanliness practices including fire hazards and the control of combustible material
- work activities being performed in accordance with approved activities
- physical security
- HP instrumentation is operable and calibrated

c. During this report period the licensee completed all scheduled outage activities and performed various evolutions to prepare the plant for operation. The NRC inspector observed the final portions of the Control Element Assembly (CEA) coupling procedure being performed in accordance with MP-RC-10-7, "Procedure for Coupling Control Element Assemblies." The NRC inspector observed control room activities during the reactor fill operation. This was done in accordance with OI-RC-2A, "Reactor Coolant Fill Instruction," and it was noted that the current revision to the procedure was being used, prerequisites were met and signed off, and the procedure signoffs were current.

The cold hydrostatic test was performed using OI-RC-2C, "Cold Hydrostatic Test of RCS," which placed the reactor coolant system in a water solid condition at the shutoff head of the low pressure safety injection pump. The NRC inspector reviewed the prerequisites and verified that shutdown cooling was in operation, that the HPSI, charging, and reactor coolant pumps were caution tagged in pull-stop, that Checklists OI-RC-2B-CL-A and OI-RC-2B-CL-B had been completed, that the RCS vent to containment atmosphere had been secured, and that RCS temperature was between 82°-130° F. The cold hydrostatic test was completed satisfactorily, and the plant continued preparations to heatup above 210° F.

OP-1, "Master Checklist for Startup or Trip Recovery," is the controlling document taking the plant from cold shutdown to hot standby condition. The NRC inspector reviewed all the primary plant checklists performed for Section 2 of OP-1 and verified that they were complete and signed off, and that independent verification was performed where required. In addition, the NRC inspector performed the following independent valve lineups:

- DG-1-CL-A, "No. 1 Diesel Generator - Starting Air System"

- DG-1-CL-B, "No. 1 Diesel Generator - F.O. System"
- DG-2-CL-A, "No. 2 Diesel Generator - Starting Air System"
- DG-2-CL-B, "No. 2 Diesel Generator - F.O. System"
- MS-2-CL-A, "Main Steam System (partial)"
- EE-1-CL-B, "4160 KV System (partial)"

No violations or deviations were identified.

3. Surveillance Testing

The NRC inspector witnessed portions of the following surveillance tests:

a. ST-RPS-12, F.2 (Monthly) and F.3 (Refueling) Axial Power Distribution Channels. Section F.3 called for Calibration Procedures CP-A/APD-1, CP-B/APD-1, CP-C/APD-1, and CP-D/APD-1 to be performed on their respective safety channel. It was verified that for both surveillance sections, the revised channel coefficient settings for cycle 9 operation as identified in Technical Services Memo TS-FC-84-110H, Table A-1 had been incorporated into the test procedures.

b. ST-RPS-4, F.3 (Refueling) Thermal Margin/Low Pressure Channels. The NRC inspector verified that revised channel coefficient settings for cycle 9 operation as identified in Technical Services Memo TS-FC-84-110H, Table A-3 had been incorporated into the test procedure.

c. ST-ISI-CC-3, F.1 (Quarterly) Component Cooling Water Pump Inservice Testing (AC-3B and AC-3C only).

In the above surveillance tests, the NRC inspector verified, where applicable that:

- testing was scheduled in accordance with Technical Specification requirements
- procedures were being followed
- calibrated test equipment was being used
- qualified personnel were performing the tests
- limiting conditions for operation were being met
- test data were being accurately recorded

No violations or deviations were identified.

4. Maintenance Activities

The NRC inspector witnessed portions of the work performed on the following maintenance items:

a. Maintenance Order (MO) 841339, "Check Raychem Splices." This MO was written to perform inspection of 54 solenoid splices to verify that they were Raychem or not. The NRC inspector observed the inspection of PCV-2949 Solenoid and noted that a Raychem splice did not exist. MO 841959 had been written to, "put RTV splices on those items which do not have Raychem," and the NRC inspector observed the work being performed by a qualified craftsman.

b. MO 841040, "Remove 1042B Check Valve Lid and Internals." The MO referenced MP-MSIV-1, "Disassembly Procedure for HCV-1041A and HCV-1042A," and the NRC inspector noted that Revision 4 was used up to the time of the steam generator helium test and that Revision 6 was present at the jobsite to cover work performed after the helium test. The NRC inspector verified that the required signoffs were current, that QC hold points were identified and observed, and that a tank entry permit was completed and filled out. It was noted that spare parts used were identified on the MO by description, purchase order request, and stock number for material accountability. The QA material conformance tags were present with the work package. The NRC inspector observed machining being done on the disc seating surface and verified that a QC hold point existed to PT the surface once cutting was complete. The taillink assembly had already been magnetic particle tested and signed off by a qualified QC inspector.

c. MO 841752, "HCV-509A Diaphragm Replacement." This work was performed using a PRC procedure, and the NRC inspector verified that QC hold points were established and observed, that the applicable Technical Specification requirements were identified, and that qualified personnel were assigned to the work. It was noted that postmaintenance testing requirements were identified and called for the performance of ST-ISI-WD-1, F.1, "Waste Disposal Valves Inservice Testing," and ST-CONT-3, "Containment Isolation Valves Leakage Rate Test - Type C." The NRC inspector reviewed the completed work procedure and noted that it was signed off properly, that RWP 252 was used on this job, and that QC personnel performed a postmaintenance review of the valve repair and witnessed the valve closeout.

d. MO 841751, "HCV-258 Repair." The NRC inspector observed this work in-progress and reviewed the maintenance package at the jobsite. It was noted that a safety evaluation (FC-154) was completed, that a separate RWP was issued to cover this work, and that procedure signoffs were current. Maintenance Procedure MP-MOV-1, "Motor Operated "Limitorque" Valves Type SMB/HMB Limit Switch and Torque Switch Replacement and Adjustment Procedure," was used, and Appendix B, Part I was filled out covering the torque switch/limit switch setting procedure. The valve opening and closing current was recorded in Appendix C to the maintenance procedure. Postmaintenance testing was performed using ST-ISI-CVCS-1, F.3, "Chemical and Volume Control Valves Inservice Testing," and the NRC inspector observed that the stroke time was 15.2 seconds; well within the maximum 46 seconds allowed.

e. MO 842407, "TIB-3B Replacement." On Monday, May 21, 1984, TIB-3B (a 4160/480V transformer) failed causing smoke in the switchgear room and actuating the halon fire suppression system. The transformer was de-energized and the licensee was able to check for faults and reenergize 480V Bus 1B3B by feeding accross through Bus Tie BT-1B3B. Within 20 minutes a notification was made to the NRC emergency duty officer because access to the switchgear room was restricted for a short time. The licensee had a spare transformer in stock and this MO was written to perform the replacement. The NRC inspector reviewed the MO and verified that it was properly signed off and approved, that appropriate tags had been assigned and hung, that QA/QC requirements were identified, and that the applicable Technical Specification reference was identified. The NRC inspector observed this work over a three-day period and noted that a PRC approved procedure was prepared to cover this task, procedure signoffs were maintained current, QA/QC hold points were observed, testing on the new transformer was completed satisfactorily, and qualified personnel were assigned to the job.

No violations or deviations were identified.

4. Followup of Steam Generator Tube Failure

On May 16, 1984, at about 4:42 p.m., the Fort Calhoun Station experienced a single tube failure in Steam Generator RC-2B while leak testing the reactor coolant system (RCS) in accordance with OI-RC-2B, "Reactor Coolant Vent and Leak Test Instruction." The plant was in hot shutdown at the end of a 2½ month refueling outage, RCS boron concentration was approximately 2100 ppm,and RCS temperature at 392° F. The pressurizer was solid, and pressure was being increased using the charging pumps to perform the leak test at 2250 psia. At about 1800 psia, pressure started to level out and an increase in RC-2B level was noted. Within ten minutes RCS pressure was reduced to 560 psia and lab technicians had drawn a steam generator sample to test for activity. Results indicated the sample to be radioactive and helped confirm a tube failure in RC-2B. The licensee initiated a cooldown using the unaffected steam generator and the atmospheric dump valve. Approximately 25 minutes into the event, the licensee declared an unusual event and notified the NRC emergency duty officer. Initial radiological surveys indicated no contamination or offsite release. Plant cooldown continued and at 8:05 p.m. shutdown cooling was initiated. Shortly after midnight on May 17, 1984, the licensee emergency organization was secured from the unusual event since the plant was in cold shutdown and notified the NRC of this action at 12:40 a.m.

The NRC inspector was in the control room observing the leak test at the time of the event, and participated in the notification phone call to the NRC emergency duty officer with the plant manager. Later that evening the NRC inspector talked with a Region IV representative to apprise him of the situation. A daily report entry was prepared by Region IV, and the licensee has described this incident in Operations Incident Report No. 1912, with an LER to be submitted within 30 days of the incident.

The NRC inspector verified that the plant was placed on shutdown cooling, depressurized, and drained in accordance with OI-SC-1, "Initiation of Shutdown Cooling," and OI-RC-5, "Reactor Coolant System Draining." Draining of the steam generator was performed using a PRC approved procedure performed under MO 842170, "Drain "B" Steam Generator Secondary Side." This procedure was used to drain the secondary side of RC-2B by pumping the contents to the Safety Injection Refueling Water Tank and to provide for boron sampling on the primary system during this evolution. On May 19, 1984, the primary manways on RC-2B were removed and the licensee determined that a single tube failure (Row 84, Line 29) had occured on the top of the generator between the scallop bars in the vertical batwing support on the hot leg side of RC-2B. A review of eddy current inspection reports for this generator revealed that the failed tube was inspected during the 1982-1983 outage and was found to have no reportable anamolies. The tube was reinspected in March 1984, and was again reported to be defect-free. Following the tube failure, the eddy current data from the 1984 inspection was reanalyzed and it was found that an error had occurred in the original interpretation. Instead of being defect-free, as was originally reported, the tube actually contained a crack-like indication with an approximate depth of 0.045 inch or 90 to 95 percent through-wall. This was considered to be a gross error and the true condition of the remainder of the tubes inspected during this outage immediately became suspect. The 1984 inspection was performed by Combustion Engineering personnel and the data was analyzed and reported by Zetec Corporation. It is common practice for eddy current inspection data to be evaluated and reported by persons other than those who actually conducted the inspection. A rereview of all of the 1984 inspection data was performed by Combustion Engineering analysts and it was found that the failed tube was the only one that was erroneously interpreted.

On May 22, 1984, the NRC inspector attended a licensee planning meeting in which it was determined that the failed tube was to be removed. Vendor personnel were brought in to accomplish this work, procedures were prepared and approved to cover this task, and arrangements were made to package and ship the failed tube to Combustion Engineering in Windsor, Connecticut for analysis. The NRC inspector attended the 7:00 a.m. briefing on May 25, 1984, as the licensee made final preparations to enter RC-2B to cutout the failed tube and one other tube for interference. The tubes had been marked by an OPPD engineer, lights and ventilation were established, radiation surveys had been performed, and two-man crews were setup. The briefing included a review of work to be performed, estimated times involved, health physics considerations and stay times, tool accountability, and handling of the pieces from RC-2B to the cask decon area for viewing. Once the cutout tubes (two segments in four pieces) were removed and placed inside a tent in the cask decon area, the NRC inspector joined OPPD and CE personnel in viewing the segments. This visual inspection showed the failure to be a longitudinal crack at the six o'clock location about 1 3/8 inches long and 1/8 inch wide. There appeared to be no thinning of the wall, fretting, or discoloration present, and the failure was characterized as, "intergranular stress corrosion cracking," by licensee observers.

During the period May 23-25, 1984, the NRR project manager for the Fort Calhoun Station was onsite to observe licensee activities and provide interface with NRC technical personnel in Washington. Extensive conference calls were held on the 24th and 25th between the licensee, Region IV, and NRR in Washington in which OPPD described the results of their eddy current test data review, current plant status, and their proposed actions to return the plant to operational status. This exchange of information was followed by a meeting between the licensee and NRC representatives at the Phillips Building in Bethesda, Maryland, on May 29, 1984. Minutes of this meeting will be issued by NRR under a separate document. The licensee's proposed plans and commitments, along with descriptions of the incident and steam generator inspections, are contained in OPPD Letter LIC-84-160 dated May 31, 1984, from W. C. Jones to J. T. Collins, Regional Administrator, USNRC, Region IV.

During the period May 26-28, 1984, a regional inspector from Region IV was onsite observing and evaluating the licensee's eddy current testing program. The NRC inspector reviewed the inspection procedures utilized for the last two inspections (1982 and 1984) and found that there were no significant differences. Both inspections were performed using the multi-frequency technique and both the differential and absolute modes. Additional tube examinations were being performed in the "B" Steam Generator at the end of this inspection period but no final evaluation of the data had been made. This additional inspection and evaluation was being performed by Combustion Engineering personnel.

At the close of this report period, the Fort Calhoun Station was in cold shutdown (Mode 5) and drained down with the primary manways removed from RC-2B. The licensee is in the process of eddy current testing all the tubes in RC-2B accessible by the remote probe insertion machine, and the analyses of the failed tube at the CE facility continues.

5. Exit Interview

The NRC inspector met with the plant manager on June 5, 1984, to summarize the scope and findings of the inspection.