APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

NRC Inspection Report: 50-285/89-44 Licensee: 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: November 1-30, 1989

Inspectors: P. Harrell, Senior Resident Inspector
T. Reis, Resident Inspector
R. Mullikin, Project Engineer
R. Farrell, Senior Resident Inspector, Fort St. Vrain

Approved: T. F. Westerman 12/21/89
T. F. Westerman, Chief, Project Section B Date
Division of Reactor Projects

Inspection Summary

Inspection Conducted November 1-30, 1989 (Report 50-285/89-44)

Areas Inspected: Routine, unannounced inspection including review of previously identified items; operational safety verification; plant tours; safety-related system walkdown; monthly maintenance observations; monthly surveillance observations; security observations; radiological protection observations; in-office review of periodic, special, and nonroutine event reports; review of onsite events; review of NRC bulletins and generic letters; and assessment of the local public document room.

Results: Of the 12 areas inspected, no violations or deviations of NRC requirements were identified.

The inspectors reviewed the areas discussed below. The discussion provides an overall evaluation of each area.

- The inspectors reviewed the actions taken by the licensee in response to previously identified items and it appeared that the licensee had appropriately implemented both short- and long-term actions to prevent recurrence of the identified problems.

- The inspectors performed numerous tours of the plant during this inspection period. During the tours, no significant problems were noted.

- During observation and review of maintenance and surveillance activities, the inspectors noted no problems with the procedures, documentation, or activities reviewed.

- In the area of security operations, the inspectors noted no problems. Significant progress continues in the security upgrade program.

- During numerous tours of the radiologically controlled area, the inspectors found no problems with the implementation of the radiological protection program.

- During review of onsite events, the inspector noted that the licensee took proactive measures in preparation for a planned outage of the 161-kV offsite power supply. The licensee prepared a contingency plan to describe the actions to be taken in the event that river level dropped. The licensee's efforts toward generation of the plan were considered to be very good.

- During review of the actions taken by the licensee with respect to requirements contained in NRC bulletins and generic letters, it appeared that the licensee's actions were adequate and addressed the required actions specified by bulletins and generic letters.

DETAILS

1. Persons Contacted

*R. Andrews, Division Manager, Quality and Environmental Affairs
*J. Bobba, Supervisor, Radiation Protection
*C. Brunnert, Supervisor, Operations Quality Assurance
*J. Chase, Manager, Nuclear Licensing and Industry Affairs
*G. Cook, Nuclear Licensing Engineer
M. Core, Supervisor, Maintenance
*S. Gambhir, Division Manager, Production Engineering
*G. Gates, Executive Assistant to the President
*C. Huang, Plant Chemist, Special Assignment
*R. Jaworski, Manager, Station Engineering
J. Kecy, Supervisor, Systems Engineering
*M. Lazar, Supervisor, Operations and Technical Training
D. Matthews, Supervisor, Station Licensing
*W. Orr, Manager, Quality Assurance and Quality Control
*G. Peterson, Manager, Fort Calhoun Station
*A. Richard, Assistant Manager, Fort Calhoun Station
J. Sefick, Manager, Security Services
*R. Short, Supervisor, Special Services
*C. Simmons, Station Licensing Engineer
*S. Swearngin, Nuclear Safety Review Group
*J. Tills, Assistant Manager, Fort Calhoun Station
D. Trausch, Supervisor, Operations
*B. Van Sant, Design Engineering, Nuclear
*S. Willrett, Manager, Administrative Services

*Denotes attendance at the monthly exit interview.

The inspectors also contacted other plant personnel, including operators, technicians, and administrative personnel.

2. Plant Status

During this inspection period, the FCS operated at 100 percent power. No safety system challenges occurred.

Due to depressed Missouri river levels compounded by river debris fouling the intake structure grids and traveling screens, the licensee experienced periodic interruptions of circulating water flow. The interruptions were readily remedied by operator action in backwashing the screens and throttling circulating water discharge. A detailed discussion of problems associated with river level is provided in paragraph 12 of this inspection report.

3. Review of Previously Identified Items (92701 and 92702)

a. (Closed) Unresolved Item 285/8815-11: Capability to establish hot-leg injection.

This unresolved item is related to questions regarding the licensee's capability to establish hot-leg injection during postaccident conditions. In 1978, Combustion Engineering recommended that the licensee provide a remote manual method for operation of four safety injection system valves to ensure that hot-leg injection could be established. The licensee did not install remote operators for the valves. This issue was forwarded to the NRC's Office of Nuclear Reactor Regulation (NRR) for review and verification that hot-leg injection could be established with the present plant configuration.

NRR reviewed the configuration of the safety-injection system and issued a safety evaluation report (SER) on January 10, 1989. The SER identified several concerns with the system installation. On June 16, 1989, the licensee provided a response to the concerns. Based on review of the licensee's submittal, NRR determined that the current safety-injection system configuration was adequate for establishing hot-leg injection. NRR's conclusions were documented in a letter dated July 27, 1989.

Based on the review performed by NRR, this item is considered closed.

b. (Closed) Open Item 285/8815-13: Adequacy of calculations issued by licensee to address a loss of containment integrity.

This open item is related to calculations performed by the licensee to verify that 10 CFR Part 100 limits would not have been exceeded in the event of a DBA due to the loss of containment integrity which occurred when a technician failed to reinstall a tube cap after calibration of the pressure switch used to monitor containment pressure.

The calculation performed to verify that Part 100 limits would not have been exceeded was transmitted to NRR for review. In a memorandum dated October 24, 1989, NRR stated that the licensee's calculation had been reviewed and it appeared that the conclusions reached by the calculation were valid.

Based on the review performed by NRR, this item is considered closed.

c. (Closed) Open Item 285/88201-12: Concern with danger tagging of equipment.

During reviews of maintenance activities performed by the Operational Safety Team Inspection (OSTI), it was found that the licensee's mechanism for danger tagging of electrical breakers provided for an unnecessary vulnerability. Specifically, danger tags were found to be placed on the breakers themselves. If a breaker was removed from its motor control center (MCC), a tag would not remain on the MCC identifying the status of the breaker. The OSTI was concerned that another breaker could be substituted in the MCC cubicle where the breaker had been removed; thereby, placing equipment and personnel at risk. The OSTI noted that Procedure SO-0-20, "Equipment Tagging Procedure," did not address this potential vulnerability and considered this a weakness.

The licensee revised Procedure SO-0-20 on September 7, 1989, to satisfactorily address this vulnerability. The inspector reviewed the procedure change and verified that operations personnel were trained on the revision. Training was provided by Hotline 89-036. Based on the procedural change and implementation of appropriate training, this item is considered closed.

d. (Closed) Violation C.1 (OSTI Unresolved Item 285/88201-15): Intent changes were made to procedures without plant review committee (PRC) approval.

This violation involved a finding by the OSTI that the licensee had made temporary changes to surveillance test procedures that were deemed to change the intent of the procedures. TS 5.8 states, in part, that temporary changes to procedures may be made provided that the intent of the original procedure is not altered and the change is documented, reviewed by the PRC, and approved by the Manager, FCS, within 14 days of implementation. Of 30 temporary changes reviewed, the OSTI identified two changes to surveillance tests where intent changes were made and the PRC did not review the changes.

The licensee admitted to the violation, as written, and attributed the root cause as improper guidance given individuals as to what constituted an intent change with respect to documentation of test results and failure to follow established procedures.

As corrective action, the licensee incorporated guidance regarding the acceptable methods to document results while performing a surveillance test in Item 60, "Improve Controls over Surveillance Test Program," of the Safety Enhancement Program (SEP). All surveillance tests are being revised as part of the actions specified in SEP Item 48, "Safety-Related Procedures Upgrade Project."

Additionally, the root cause aspect of failure to follow procedures has repeatedly been addressed by licensee management. The licensee has stressed to its employees that procedural compliance is mandatory. The inspector has noted that instances of procedural noncompliance are declining.

Based on the incorporation of the corrective actions of this violation into the SEP and licensee efforts in the procedural compliance area, this violation is considered closed.

e. (Closed) Violation F.1 (OSTI Unresolved Item 275/88201-17): Inadequate corrective action program.

During stroke time testing of Valve HCV-1749, the stroke time was measured locally as 12.2 seconds and remotely as 3.8 seconds. No effort was made by the licensee to resolve this discrepancy between the stroke times.

The licensee revised Procedure SO-G-23, "Surveillance Test Program," to require personnel performing surveillance tests to immediately report all anomalies and deficiencies to their immediate supervisor. The actions taken by the licensee should prevent future delays in initiating corrective action for discrepant conditions.

f. (Closed) Unresolved Item 285/88201-18: Failure to measure remote indication for valves in the ASME Code, Section XI program.

During a review of the surveillance testing program performed by the OSTI, it was noted that Procedure ST-ISI-FW-1, "Feedwater Valves Inservice Testing," did not contain instructions for verifying the remote position indication of Valves HCV-1107B and HCV-1108B. Verification of remote indication is required, at a minimum of 2-year intervals, by Subsection IWV-3300 of Section XI.

The licensee provided a response that stated that this problem was self-identified and documented it in a report entitled "Verification of Inservice Testing of ASME Category A, B, and C Valves," dated December 31, 1987. This assertion was verified as accurate.

Corrective actions taken to address this item, as well as other surveillance testing deficiencies found by the OSTI, were incorporated into the SEP. The inspector reviewed the action items required to close SEP Item 72 and found this concern specifically addressed. SEP Item 72 has been completed, with the exception of formal issuance of the revised inservice inspection (ISI) procedures. The ISI procedures will be issued in conjunction with SEP Item 48. This concern is closed based on incorporation of the appropriate actions into the SEP program.

g. (Closed) Violation C.2 (OSTI Unresolved Item 285/88201-19): Temporary procedure changes not reviewed by the PRC and approved by the plant manager within 14 days.

This item involved the use of Revision 25 of Procedure ST-CONT-2, "Local Leak Rate Testing-Type B," in the field by a technician. Three on-the-spot changes (OTSC) were attached to the procedure being used by the technician. The OSTI inspector noted that the current revision of Procedure ST-CONT-2 was Revision 29 and was concerned that the technician was not using an appropriately approved copy of the procedure. This item also involved the licensee's control over implemented OTSCs that may later be identified as incorrect.

In response to this item, the licensee stated that Revision 26, with the three appended OTSCs, contained the same requirements that were provided in Revision 29 of the procedure. The licensee stated that the use of OSTCs was an appropriate method for changing procedural requirements. In addition, the licensee stated that the three OTSCs appended to Procedure ST-CONT-2 had been reviewed and approved by the PRC and plant manager.

The inspector reviewed the actions taken by the licensee to address this item. The inspector also reviewed Procedure SO-G-30, "Setpoint/Procedure Changes and Generation," and it appeared that the appropriately administrative controls were established to address the use of OTSCs. No problems were noted during the reviews.

h. (Closed) Violation D.2 (OSTI Unresolved Item 285/88201-21): Inadequate instructions for performance of safety-related activities.

   This violation was related to inadequacies identified with Procedure ST-ISI-RW-1, "Raw Water Valves Inservice Testing." The procedure did not provide instructions for testing of Bettis valve operators, but had been issued for testing stroke times of valves with Bettis operators.

   The inspector reviewed Revision 12 of Procedure ST-ISI-RW-1, which resulted in the violation. The inspector also reviewed the current Revision 18 of the procedure.

   Revision 18 appeared to correctly address testing of the subject valves.

i. (Open) Unresolved Item 285/8909-04: Seismic qualification of a fire water line over Emergency Diesel Generator (EDG) 1.

   The licensee prepared a calculation to verify the seismic qualification of the fire water line directly over EDG 1. The inspector found two supports for this line removed and had questioned the seismic qualification of the line. The calculation was generated to verify that removing the supports did not invalidate the seismic qualification of the fire water line, thus making EDG 1 potentially inoperable.

   The licensee's calculation (FC-05009) was reviewed by NRR. NRR issued a memorandum, dated September 26, 1989, noting apparent inadequacies in the calculation. NRR's comments are listed below and reference the specific page number of Calculation FC-05009:

   (1) Page 3. Use of ASME Code Case N-411 damping values is not acceptable because it does not meet the conditions specified in Revision 24 of Regulatory Guide 1.84.

(2) Page 3. It is not clear how the multiplication factor of 1.5, used to estimate the maximum vertical response acceleration at 5 percent damping curve from that of 7 percent damping curve, was derived.

(3) Page 9. It is not clear how the calculation of maximum bending stress, fb = 6541 psi, was calculated.

(4) Page 10. It is not clear how the maximum bending stress, fb = 19,166 psi, was calculated.

(5) Page 11. An increase of one-third of allowable stress due to seismic load is questionable.

The licensee stated that contact with NRR would be made in the near future to discuss the apparent inadequacies of Calculation FC-05009. This item remains unresolved pending submittal of the information by the licensee and review of the information by NRR.

j. (Open) Deviation 285/8909-06: Testing of the deluge valves for the fire water curtains.

This deviation involved the licensee's failure to test the deluge valves for the fire water curtains installed over the doors between the auxiliary and turbine buildings. The requirement for deluge valve testing was established by the NRC by issuance of an SER, dated August 23, 1978.

In response to this deviation, the licensee submitted a letter, dated July 27, 1989, to the NRC requesting that the deluge valves be removed from the SER. The licensee submitted additional information concerning the request for deluge valve removal from the SER on September 15, 1989. In subsequent discussions between NRR and the licensee, NRR stated that the deluge valves would not be removed from the SER and that the licensee was required to test the valves. NRR's conclusion, based on the installation and testing of the valves, was that the licensee was required to meet the requirements of Appendix R to 10 CFR Part 50. Based on the denial of the licensee's request to remove the deluge valves from the SER, the information submitted in response to this deviation has become outdated. The inspector discussed the status of the response with the licensee. Licensee management stated that a revised response would be submitted in the near future.

This item remains open pending receipt of the revised response, review and acceptance of the response by the NRC, and review of the actions taken by the licensee to address the problems identified by this deviation.

k. (Open) Violation 285/8913-01: This violation involved two examples of the failure to follow procedures.

The first example involved the licensee's failure to properly implement Procedure SO-O-20, "Equipment Tagging Procedure." A danger tag was installed on Valve MS-100 and the valve was not in the required position. The danger tag stated that the valve position was shut; however, the valve was found to be open.

The licensee issued Incident Report 890403 on March 23, 1989, to review the root cause and develop corrective actions. The licensee determined the cause to be personnel error. Training Hotline 89-015 was provided to all licensed and nonlicensed operators, shift technical advisors, and operations training instructors. This hotline, issued March 29, 1989, provided information on this event occurrence and emphasized the importance of ensuring that valve positions are consistent with the directions on danger tags.

The inspector reviewed Hotline 89-015 and the associated training records. The licensee's corrective action appeared to adequately address this concern.

The corrective action on the second example of the failure to follow procedures with respect to installation of cable tray covers remains open. The licensee is currently implementing corrective actions to address this specific problem. Review of the licensee's actions will be performed in the near future.

l. (Open) Open Item 285/8913-02: Seismic qualification of Valve HCV-1388B.

This item was related to questions regarding the seismic installation of Valve HCV-1388B. The licensee submitted calculations to the inspector for verification of the seismic installation of the valve. The calculation was forwarded to NRR for review.

NRR reviewed the calculation and noted that the calculation was incomplete. NRR could not determine, by reviewing the calculation, whether or not the valve installation was appropriate due to the inadequate information contained in the calculation.

The inspector notified the licensee of the apparent incomplete calculation. Licensee management stated that NRR would be contacted in the near future to discuss the concerns with the calculation. This item remains open pending receipt and review of the revised calculation by the NRC.

m. (Closed) Deviation 285/8926-01: Steam lines not supported in accordance with code requirements.

This deviation identified 1-inch steam drain lines not being supported per USAS B31.1-1967. The licensee committed to this code in Appendix A of the Updated Safety Analysis Report.

In a letter dated August 24, 1989, the licensee denied the deviation and provided an analysis that verified that the lines were adequately supported. The licensee stated that the support spacings provided by the code are recommendations only.

Although the inspector agreed with the licensee's assertion that the code dimensions are recommendations only, the introduction to the code states that, where latitude in design is taken, the designer is responsible for demonstrating the validity of his approach. The licensee could not provide a preexisting analysis demonstrating validity, but subsequently verified the adequacy of the 1-inch stream drain line supports by analysis. The inspector has examined the licensee's analysis of the existing support spans and found them satisfactory; therefore, further action is not required. Acknowledgement and acceptance of the denial of the deviation was transmitted to the licensee in a letter from Region IV dated November 14, 1989.

No violations or deviations were identified.

4. Operational Safety Verification (71707)

The inspectors conducted reviews and observations of selected activities to verify that facility operations were performed in conformance with the requirements established under 10 CFR, the licensee's administrative procedures, and the Technical Specifications (TS). The inspectors made several control room observations to verify the following:

- Proper shift staffing was maintained and conduct of control room personnel was appropriate.
- Operator adherence to approved procedures and TS requirements was evident.
- Operability of reactor protective system, engineered safeguards equipment, and the safety parameter display system was maintained. If not, the appropriate TS limiting condition for operation (LCO) was met.
- Logs, records, recorder traces, annunciators, panel indications, and switch positions complied with the appropriate requirements.
- Proper return to service of components was performed.
- Maintenance work orders (MWO) were initiated for equipment in need of maintenance.
- Management personnel toured the control room on a regular basis.
- Control room access was properly controlled.

° Control room annunciator status was reviewed to verify operator awareness of plant conditions.

° Mechanical and electrical temporary modification logs were properly maintained.

° Engineered safeguards systems were properly aligned for the specific plant condition.

During review of this area, the inspectors identified the following items:

a. On November 3, 1989, the licensee requested an extension for response to the weaknesses identified in NRC Inspection Report 50-285/89-29. The report documented the results of the annual emergency exercise held in July 1989. The request was for an extension of the issuance of a response to the exercise weaknesses from November 24, 1989, to January 12, 1990. The reason given by the licensee was that the additional time would be used to perform an extensive root cause analysis and evaluation.

   On November 6, 1989, the Chief, Project Section B, Division of Reactor Projects, Region IV, in conjunction with the Chief, Safeguards and Emergency Planning Section, Division of Radiation Safety and Safeguards, Region IV, authorized the licensee's request for an extension to January 12, 1990. The licensee was immediately notified of the approved extension by the inspector.

b. During this inspection period, discussions were held on site between the licensee and the NRC on what actions are required if the licensee entered TS 2.0.1. This TS requires that the plant be placed in hot shutdown within 6 hours. Once entry into TS 2.0.1 has been made, the TS does not provide a 1-hour grace period before plant shutdown must be initiated as is found in the newer TS version of TS 3.04. Based on discussions between Region IV and NRR Project Directorate IV, the following guidance was provided:

   ° Since no grace period exists in TS 2.0.1, as currently written, entry in TS 2.0.1 constitutes initiation of a plant shutdown as discussed in Section 50.72(b)(i)(A).

   ° Section 50.72(b)(i)(A) requires that a 1-hour report be made due to the initiation of any plant shutdown required by the plant's TS. Therefore, a 1-hour report is required. The 30-day licensee event report requirements of Section 50.73(a) are also applicable.

     It is recognized that the licensee must take actions (e.g., notification of the electrical dispatcher, calculation of the amount of boric acid to be injected into the reactor coolant system for initiation of the shutdown, etc.) before plant power can be reduced. Therefore, an immediate power reduction is not required. However, it is anticipated that these actions will be promptly initiated after entry into TS 2.0.1 and that shutdown will be completed within the 6 hours.

c. During this inspection period, a survey was conducted by the Region IV office regarding the code safety valves installed at the FCS. The results of the survey are provided below:

(1) Code safety valve vendor

° Pressurizer Codes Safeties (PCS): Crosby, Style HV-BP-86, Type E

° Main Steam Safety Valves (MSSV): Dresser Maxiflow

(2) Loop seals used with valves

° PCS: Yes

° MSSV: No

(3) Type of valve testing

° PCS: Each refueling outage, no on-line testing done

° MSSV: Each refueling outage, no on-line testing done

(4) Who performs testing program

° PCS: Removed and sent to Wyle Laboratories for testing. The valves are not tested by the vendor using a loop seal.

° MSSV: Done in place by licensee using Hydroset method. Removed and sent to Wyle Laboratories for refurbishment and setting if out of specification.

d. NRC Inspection Reports 50-285/89-23, 50-285/89-32, and 50-285/89-33 provided discussions of degraded raw water (RW) pumps and fouling of the RW/component cooling water (CCW) heat exchangers that resulted in a condition where insufficient flow was available to provide cooling for the design basis accident (DBA) at elevated river temperatures with less than four CCW heat exchanges in service. The licensee instituted administrative controls to maintain four heat exchangers on line and routinely verify system flow rate. The licensee agreed, via discussions with Region IV and NRR, to notify the inspector whenever one of the heat exchangers was taken off line in order that an independent assessment of cooling capability could be made.

Additionally, on September 11, 1989, RW Pump AC-10C failed to provide sufficient flow during surveillance testing to meet its acceptance criteria. RW Pump AC-10A, on the same emergency bus as Pump AC-10C, satisfactorily passed its surveillance testing. Together, Pumps AC-10A and AC-10C provided sufficient flow to meet DBA flow requirements. The licensee justified that Pump AC-10C satisfactorily met the requirements of Section IWP-3230(c) of Section XI of the ASME Code and was, therefore, operable.

The licensee implemented administrative controls to establish that Pump AC-10C was only considered operable provided Pump AC-10A was operable. Therefore, the licensee would enter the TS LCO for two inoperable pumps if Pump AC-10A became inoperable.

The licensee's analysis concluded that this approach satisfied the DBA cooling capabilities and their commitment to Section XI of the ASME Code. This approach was agreed upon by the licensee, Region IV, and NRR in a phone conference on September 20, 1989.

Operation Memorandum 89-02, "Operability of Raw Water Pumps" was approved by the PRC and issued on September 19, 1989, to implement the administrative controls.

By mid-September, the licensee had made sufficient progress in the heat exchanger sparging program and river temperatures had declined sufficiently such that DBA flow could be maintained with three of four RW/CCW heat exchangers in service. During September 20-27, 1989, RW/CCW Heat Exchanger AC-1B, the one with the highest flow resistance, was taken out of service, cleaned, and returned to service. Subsequent operation found that any combination of two pumps, on the same emergency bus, through three heat exchangers could provide DBA cooling flow. Administrative controls to maintain four heat exchangers in operation were no longer necessary.

Monthly surveillance testing of the RW Pumps AC-10A, -B, -C, and -D was again performed on October 20, 1989, and with the decreased system resistance due to cleaning of RW/CCW Heat Exchanger AC-1B, all four pumps satisfactorily met their acceptance criteria. The administrative controls implemented by Operations Memorandum 89-02 were no longer necessary.

On October 23, 1989, the licensee expressed their desire to remove both administrative controls which had been agreed upon by licensee management, Region IV, and NRR. The issues were discussed with Region IV and NRR, who concurred with the removal of the administrative controls for RW Pumps AC-10A and -C operability and maintaining four RW/CCW heat exchangers in service.

On November 1, 1989, the licensee formerly lifted the administrative controls by action of Memorandum FC-1987-89, "Recommendation for Closure of Ops Memo 89-02." The licensee has committed to address permanent corrective action of RW system flow obstruction in response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

No violations or deviations were identified.

5. <u>Plant Tours (71707)</u>

The inspectors conducted plant tours at various times to assess plant and equipment conditions. The following items were observed during the tours:

- General plant conditions, including operability of standby equipment, were satisfactory.
- Equipment was being maintained in proper condition, without fluid leaks and excessive vibration.
- Valves and/or switches for safety-related systems were in the proper position.
- Plant housekeeping and cleanliness practices were observed, including no fire hazards and the control of combustible material.
- Performance of work activities was in accordance with approved procedures.
- Portable gas cylinders were properly stored to prevent possible missile hazards.
- Tag out of equipment was performed properly.
- Management personnel toured the operating spaces on a regular basis.

During a tour of the plant on November 16, 1989, the inspector noted that the fire-barrier doors (1007-37 and 1007-38) for the EDG rooms would not fully shut each time an individual passed through the doors. Door 1007-37 provides the entry path into EDG 2 room and Door 1007-36 into EDG 1 room. The inspector notified the licensee that the door closing mechanism probably needed adjustment.

Upon notification by the inspector, the licensee added the doors to the hourly fire watch patrol list as a conservative measure. The licensee subsequently adjusted the closing mechanism on the doors and verified proper operability. The inspector reviewed the actions taken by the licensee. The actions appeared to be adequate.

The results of the plant tours performed by the inspectors indicated that the licensee is providing adequate attention to the physical condition of the plant. Work continues on painting and clean up of the plant to improve the overall appearance. Except for a few areas, plant housekeeping has been very good.

No violations or deviations were identified.

6. Safety-Related System Walkdown (71710)

The inspectors walked down accessible portions of the CCW system to verify system operability. Operability was determined by verification of selected valve and switch positions. The system was walked down using Drawing 11405-M-40, Sheet 1, Revision 23; Sheet 2, Revision 16; Sheet 3, Revision 15; and Procedure OI-CC-1, "Component Cooling," Revision 38.

During the walkdown, the inspectors noted that component identification was greatly improved. The licensee's labeling and color coding project of the CCW system is essentially complete. The project has resulted in detailed, readily identifiable and accurate labeling which will benefit the operations staff. The inspector examined all accessible components outside of containment and noted no visible deficiencies in system integrity. System appearance and identification has greatly improved since this walkdown was last performed in September 1986.

No violations or deviations were identified.

7. Monthly Maintenance Observations (62703)

The inspectors observed selected station maintenance activities on safety-related systems and components to verify that maintenance was conducted in accordance with approved procedures, regulatory requirements, and the TS. The following items were considered during the observations:

- The TS LCOs were met while systems or components were removed from service.
- Approvals were obtained prior to initiating the work.
- Activities were accomplished using approved MWOs and were inspected, as applicable.
- Functional testing and/or calibrations were performed prior to returning components or systems to service.
- Quality control records were maintained.
- Activities were accomplished by qualified personnel.
- Parts and materials used were properly certified.
- Radiological and fire prevention controls were implemented.

The inspectors observed the following maintenance activities:

- Erection of scaffolding over EDG 1 (MWO 894170)
- Repair of leaks on the air supply lines for Valves LCV-388-1 and LCV-383-2 (MWO 897590)

- Installation of a manometer for individual intake structure cell indications (MWO 895452)

- Installation of valve handwheel linkage retaining devices (MWR 4408)

A discussion of each item is provided below:

a. On November 22, 1989, the inspector observed the start of the installation of scaffolding over EDG 1. The scaffolding was being installed, in accordance with MWO 894170, so that the insulation can be replaced on the exhaust line for EDG 1.

   Prior to installation of the scaffolding, the licensee performed an analysis to verify that the scaffolding would be seismically installed once the installation was completed. The analysis was documented in Modification Request 89-003, "DG-1 Exhaust Elbow Scaffolding." To ensure that the EDG was protected during the installation of the scaffolding, the licensee identified vital components and covered the components with wooden enclosures. The enclosures were designed to prevent dropped or falling scaffolding pieces from damaging the components. The licensee also provided an erection plan for the scaffolding that ensured that access to EDG 1 by operations personnel was maintained in the event that local engine operation was required.

   The inspector reviewed the scaffolding installation activities on various occasions and noted that it appeared that the installation was being performed in accordance with the instructions provided by the MWO.

b. On November 6, 1989, the inspector was notified that the air supply system for the safety injection and refueling water tank (SIRWT) outlet valves (LCV-383-1 and LCV-383-2) were leaking slightly (i.e., as determined by soap-bubble test). The leakage was discovered during performance of the quarterly surveillance test on the valves. This test, in addition to measuring the valve stroke time, required that the fittings for the air supply line be checked for leakage.

   MWO 897590, an emergency MWO, was issued to repair the leaking fittings. The inspector observed a weakness in the MWO instructions in that it allowed corrective action to be taken on both valves simultaneously. Allowing corrective maintenance to be performed on both valves that are redundant safety equipment had the potential for rendering both valves inoperable. In actuality though, the inspector observed portions of the corrective action and only one valve was repaired at a time. The inspector noted, during subsequent interviews with the shift supervisor (SS), who released the components for work, that the SS was well aware of this vulnerability and stated that explicit verbal instructions were provided to the technicians performing the work to only disconnect the air supply lines from one valve at a time. Further, the SS indicated that he would not have allowed work instructions for both valves to be issued on a single MWO if the work had not have been of an emergency nature. The leaking joints were repaired and tested, and the components returned to service within 3 hours.

The inspector noted that the licensee's procedures did not address issuing an MWO that allowed work on redundant safety equipment. The licensee agreed to place further emphasis on the need to allow work on only one redundant component at a time. A scheduled revision to Procedure SO-M-101, "Maintenance Work Control," to be issued January 15, 1990, will provide additional guidance on the use of MWOs. Verification of the completion of this procedural enhancement by the licensee is considered an open item. (285/8944-01)

c. On November 21, 1989, the inspector observed craftsmen install a manometer to measure level in intake structure Bays A, B, and C. The manometer was installed as Temporary Modification TM-89-M-049, "Intake Structure Bay Level Indicator," as directed by MWO 895452. The purpose of the modification was to improve level indication readings in the intake structure pump bays.

Currently, depressed Missouri river levels are a concern to pump operability in the intake structure. The level in the intake structure bays can be less than the river level due to differential pressure across the intake grids and screens. Permanently installed instrumentation is only capable of measuring 24 inches of differential level before pegging high. The modification that added the temporary level indicators will supplement the existing indication by providing higher differential level readings.

The inspector found the temporary modification to be technically adequate, properly documented, and installed in accordance with approved instructions. No problems were noted.

d. On November 9, 1989, the inspector reviewed the installation of valve handwheel linkage retaining devices on Valves HCV-489A, HCV-489B, HCV-490A, HCV-490B, HCV-491A, HCV-491B, HCV-492A, and HCV-492B. The retainers were installed in accordance with MWR 4408.

The valves listed above were designed with two mechanical linkages. One linkage is used to connect the valve operator to the valve, and the other linkage is designed to connect the handwheel to the valve. Normally, the linkage for connecting the valve operator to the valve is installed and the handwheel linkage is disconnected. Due to the design of the linkage, the possibility exists that the linkage for the handwheel could inadvertently become engaged when the valve stem is rotated by the valve operator. If both linkages become engaged, the valve will not be able to move since the valve operator cannot supply sufficient force to rotate the handwheel and its installed gear box.

The inspector identified this potential vulnerability and notified the licensee. The licensee installed a valve handwheel linkage retaining device to ensure that the linkage for the handwheel would not become inadvertently engaged.

During review of other valves installed in the plant for this apparent vulnerability, the inspector noted a problem on November 13, 1989, with the mechanical linkage used to connect either the valve operator or handwheel for SIRWT Outlet Valve LCV-383-2. This problem was identical to the problem discussed above.

The inspector notified the licensee of the concern with Valve LCV-383-2. Prior to the end of this inspection period, the licensee had not taken action to install a valve handwheel linkage retaining device to prevent inadvertent engagement of the handwheel linkage. At the exit meeting, the licensee stated a linkage retaining device would be installed in the near future. This item remains open pending completion of actions by the licensee to secure the handwheel linkage. (285/8944-02)

Upon notification of the vulnerability with LCV-383-2 by the inspector, the licensee surveyed other valves installed in the plant to determine if this vulnerability existed. As a result of the survey, the licensee identified two other valves (HCV-840A and HCV-840B) where the handwheel linkage required securing. The valves are located in containment and licensee management stated that the linkage would be secured during the next refueling outage. This item remains open pending completion of securing the linkage for Valves HCV-840A and HCV-840B. (285/8944-03)

During observation of the maintenance activities performed by licensee personnel, the inspectors observed that the maintenance evolutions were performed in accordance with the appropriate procedures, as written. The inspectors also noted that the crafts performed their duties in an adequate manner.

No violations or deviations were identified.

8. <u>Monthly Surveillance Observations (61726)</u>

The inspectors observed selected portions of the performance of TS-required surveillance testing on safety-related systems and components. The inspectors verified the following items during the testing:

- Testing was performed by qualified personnel using approved procedures.
- Test instrumentation was calibrated.
- The TS LCOs were met.

- Removal and restoration of the affected system and/or component were accomplished.

- Test results conformed with TS and procedure requirements.

- Test results were reviewed by personnel other than the individual directing the test.

- Deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

- Test was performed on schedule and complied with the TS required frequency.

The inspectors observed the following surveillance test activities. The procedures used for the test activities are noted in parenthesis:

- Stroke testing of safety-related valves (ST-ISI-SI-1)

- Quarterly engineered safeguards mechanical checks (ST-SI/CS-1)

- Monthly inservice testing of RW pumps (ST-ISI-RW-3)

- Monthly testing of the station batteries (ST-DC-1)

A discussion of each surveillance observed is provided below:

a. On November 27, 1989, the inspector observed the performance of stroke timing of Valves HCV-2983, HCV-385, and HCV-386 from the control room. The stroke timing was performed in accordance with Procedure ST-ISI-SI-1, "Safety-Injection Valves Inservice Testing."

   During observations of the testing, the inspector noted that operations personnel used a calibrated stop watch; followed the procedure, as written; and successfully completed the testing to verify valve operability. No problems were noted during the observations.

b. On November 6, 1989, the inspector observed portions of the performance of Procedure ST-SI/CS-1, "Quarterly Engineered Safeguards Mechanical Checks." A portion of this test verified that SIRWT Valves LCV-383-1 and LCV-383-2 were capable of performing their intended safety function. The valves are air-operated, fail-open, air-to-shut valves. Although the test verified proper functioning of the valves, air leaks were detected in the air operating system. If the leaks were severe, the leakage could prevent the valves from being held closed. Maintaining the valves in the closed position is required by the licensee's accident analysis. The SS made the decision to initiate an emergency MWO to repair the air leaks. The air systems were repaired, postmaintenance testing was performed, and the surveillance test resumed within 3 hours. No problems were noted with the performance of the test. A concern was addressed with the instructions provided for repair of the air lines. This is discussed in paragraph 7.b of this report.

c. On November 20, 1989, the inspector observed portions of the performance of Procedure ST-ISI-RW-3, "Raw Water Pump Inservice Inspection." This is a monthly test that verifies individual pump performance and satisfies the licensee's commitment to ASME Code Section XI testing of the pumps. The test was properly performed, as written. Due to earlier RW pump performance problems, documented in NRC Inspection Report 50-285/89-39, the system engineer monitored pump performance throughout the test to readily identify any potential problems. No problems were encountered.

   The discharge flow rate measured from all pumps was found to have increased from the previous month. This was attributed to reduced system flow restrictions as a result of the licensee's efforts in disassembling and cleaning of a RW/CCW heat exchanger. The system engineer determined that the results satisfactorily fulfilled ASME Code Section XI requirements. No problems were noted.

d. During this inspection period, the inspector reviewed licensee Procedure ST-DC-1, "Monthly Testing of the Station Batteries," to evaluate conformance with Industry Standard IEEE-450-1980. With respect to addition of distilled water to individual cells, the licensee's procedure was found to conform to IEEE-450-1980 in that water was added after specific gravity was measured. If test results warranted an equalization charge, water could be added prior to the charge. In this case, the procedure directed that the equalizing charge be maintained for a minimum of 70 hours after which the cell would be placed on float charge for an additional 72 hours prior to remeasuring specific gravity. This approach was verified to conform to the IEEE Standard.

e. The inspector performed a review of the appropriate surveillance tests to verify that the program adequately addressed testing of fire detectors. The review included evaluation of the testing performed for the fire detectors located in Fire Area Zones 35 and 36. The zones designate the fire detection areas for the emergency diesel generator rooms.

Based on the observations made by the inspectors, it appeared that the licensee was adequately implementing an effective surveillance testing program. In each test observed, the inspectors noted that licensee personnel were performing the testing evolutions in accordance with the appropriate procedure, as written.

No violations or deviations were identified.

9. Security Observations (71707)

The inspectors verified that the physical security plan was being implemented by selected observation of the following items:

- The security organization was properly manned.
- Personnel within the protected area (PA) displayed their identification badges.
- Vehicles were properly authorized, searched, and escorted or controlled within the PA.
- Persons and packages were properly cleared and checked before entry into the PA was permitted.
- The effectiveness of the security program was maintained when security equipment failure or impairment required compensatory measures to be employed.
- The PA barrier was maintained and the isolation zone kept free of transient material.
- The vital area barriers were maintained and not compromised by breaches or weaknesses.
- Illumination in the PA was adequate to observe the appropriate areas at night.
- Security monitors at the secondary and central alarm stations were functioning properly for assessment of possible intrusions.

It appeared, based on the observations made by the inspectors, that the licensee's guard force was adequately performing its duties. The security system is currently being extensively modified and the extent of the modifications require that extensive compensatory measures be taken. The inspectors noted that the compensatory measures have been very good and compensate for all security system degradations.

No violations or deviations were identified.

10. Radiological Protection Observations (71707)

The inspectors verified that selected activities of the licensee's radiological protection program were implemented in conformance with the facility policies and procedures and in compliance with regulatory requirements. The activities listed below were observed and/or reviewed:

- Health physics (HP) supervisory personnel conducted plant tours to check on activities in progress.

- HP technicians were using calibrated instrumentation.

- Radiation work permits contained the appropriate information to ensure that work was performed in a safe and controlled manner.

- Personnel in radiation controlled areas (RCA) were wearing the required personnel monitoring equipment and protective clothing and were properly frisked prior to exiting an RCA.

- Radiation and/or contaminated areas were properly posted and controlled based on the activity levels within the area.

Based on the observations and reviews performed by the inspectors, it appeared that the licensee was implementing an effective radiological protection program.

No violations or deviations were identified.

11. <u>In-Office Review of Periodic, Special, and Nonroutine Event Reports (90712 and 90713)</u>

In-office review of periodic, special, and nonroutine event reports was performed by the inspectors to verify the following, as appropriate:

- Correspondence included the information required by appropriate NRC requirements.

- Test results and supporting information were consistent with design predictions and specifications.

- Planned corrective actions were adequate for resolution of identified problems.

- Whether or not any information contained in the correspondence report should be classified as an abnormal occurrence or additional reactive inspection is warranted.

- Correspondence did not contain incorrect, inadequate, or incomplete information.

The inspectors reviewed the following correspondence:

- Clarification of Design Aspects of New Control Room Charcoal Filters, dated October 30, 1989

- Decommissioning Funding Plan for the FCS, dated October 27, 1989

- Special Report on Inoperability of Fire Barriers, dated October 27, 1989

- Proposed Program Response to Generic Letter 88-20, Supplement 1, dated October 31, 1989
- Acceptance of Amendment to Indemnity Agreement, dated November 1, 1989
- Response to Generic Letter 89-07, "Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs," dated November 6, 1989
- October Monthly Operating Report, November 14, 1989
- Operating Report for October 1989, undated

During review of reports, personnel identified 10 CFR Part 21 reports submitted by suppliers or vendors that appeared to be applicable to the licensee's facility. The resident inspector provided a copy of a letter dated September 29, 1989, from the Limitorque Corporation related to cam-type torque switches in Model SMB-000 and SMB-00 valves (RIV Reference Number 89-12) to the plant licensing engineer for review of applicability by the licensee.

No violations or deviations were identified.

12. Review of Onsite Events (93702)

During this inspection period, the inspectors reviewed the following events:

a. On November 21, 1989, the licensee removed the 161-kV offsite power supply from service. The line was removed from service to replace a power pole that had shown indications of deterioration. The 161-kV power supply represents one of two available offsite power supplies. The other source is the 345-kV supply that supplies loads via a 345/22-kV transformer. In the event that the 345-kV supply is also lost, it would place the plant in a natural circulation mode of operation since no power would be available for the reactor coolant pumps.

   Prior to removal of the 161-kV supply, the licensee issued Procedure OI-EE-1B, "Transfer of Inhouse Service Between 161-kV and 22-kV," to provide the operations staff with the appropriate precautions to be taken, the instructions on how to perform the task, and the postmaintenance testing to be performed after restoring the 161-kV supply to service.

   The precautions specified by Procedure OI-EE-1B included instructions to ensure that no work is performed that could affect the stability of the electrical system; review of EOP-02, "Loss of Offsite Power/Loss of Forced Circulation;" and no maintenance or testing of the EDGs. The postmaintenance instructions provided by

Procedure OI-EE-1B required that the fast transfer function, from the 161- to the 22-kV source, be verified to operate normally.

The inspector reviewed Procedure OI-EE-1B and noted that it appeared to be adequate to address removal and restoration of the 161-kV supply. No problems were noted during the review.

b. During the past years, the local area has experienced a drought. Due to drought conditions, the flow of water into the Missouri river has been reduced by the Army Corps of Engineers (Corps) to conserve water in the upstream Gavins Point dam and reservoir. The reduced water flow in the river has caused the river level to drop. The drop in the river level presents a potential problem with the operation of the FCS since the river is the only source of the ultimate heat sink.

On November 4, 1989, the Corps reduced the flow rate from Gavins Point from 12,000 to 10,000 cubic feet per second (cfs). The reduction in flow rate resulted in the river level dropping to its current level of approximately 983 feet (mean sea level) at the FCS. Since the flow rate was reduced, the river level has been stable at this value.

Because the Missouri river is the source of the ultimate heat sink, a drop in level potentially affects the operation of the circulating water, fire protection, and RW systems. The pumps in these systems take suction from the intake structure bays that are supplied by the river. The operability of the three systems is vital to ensure that the plant can be safely shut down and maintained shut down.

TS 2.16 addresses the river level. TS 2.16 states that, when river level reaches 980 feet, a continuous watch will be posted to observe for a sudden loss of level. TS 2.16 also states that, at a river level of 976 feet 9 inches, the plant will be placed in cold shutdown using normal operating procedures. This TS is intended to ensure that the plant could be safely shutdown in the event that the river level (i.e., the ultimate heat sink) decreased.

Although unlikely the river level will drop any further due to the reduction of flow rate, the licensee has taken proactive measures to supplement the requirements stated in the TS. The licensee has established a hot line with the Corps to notify the Corps if river level starts to drop. The Corps has agreed to increase the flow rate to increase river level. In addition to establishing a method to increase river level, if required, the licensee issued a plan to describe the actions that will be taken if river level continues to drop. The plan describes the actions that will be taken at designated decreasing river levels to ensure that the plant will be safely shut down and maintained shut down. The licensee issued the plan to the operations staff in Operations Memorandum 89-04 as an interim measure to ensure that the staff had been informed of the actions to be taken on a decreasing river level. The licensee plans on revising Procedure AOP-1, "Acts of Nature," to include the information contained in the plan in the near future.

The inspectors, Region IV, and NRR have reviewed the contingency plan issued by the licensee. During review by these personnel, no problems were noted. It appeared that the licensee was proactive in addressing the problems associated with the potential loss of river level.

During the preparation of the contingency plan, the licensee requested guidance regarding the intent of TS 2.16 with respect to maintaining a continuous watch of the river level at 980 feet. The licensee indicated their intent to use the remote readout (LI-1900) in the control room as a means of complying with TS 2.16 in lieu of posting an individual outside to observe river level.

Following discussions with personnel in Region IV and NRR, the inspector indicated to the licensee that the continuous monitoring of the LI-1900 readout for a decreasing trend appeared satisfactory for complying with the continuous watch requirement in TS 2.16. However, the licensee was informed that LI-1900 could not be used as an indication of actual river level since the accuracy of LI-1900 varies between plus and minus 3 inches. If a downward river level trend is indicated on LI-1900, the licensee will measure the river level locally to determine the actual level.

No violations or deviations were identified.

13. Review of NRC Bulletins and Generic Letters (92701)

The inspector reviewed the documentation listed below to verify that the licensee had taken the appropriate actions in response to the identified issues:

a. NRC Bulletin 88-03, "Inadequate Latch Engagement in HFA-Type Latching Relays Manufactured by General Electric Company."

   The scope of Bulletin 88-03 was GE HFA relays, Types 51B, 54, 71B, 74, 151B, 154, 154B, 154E, 171B, and 174. The suspect GE relays would energize as designed, but could unlatch prematurely due to insufficient latch engagement. The licensee performed a review and documented, in a letter dated December 7, 1988, that the HFA relays in use at the FCS were not within the scope of Bulletin 88-03. This bulletin is considered closed.

b. Generic Letter 85-02, "Staff Recommended Actions Stemming from NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity."

   NRC analyses indicated that the probability of core melt from events involving steam generator tube ruptures was not a major contributor to total core melt risk, that steam generator tube ruptures are an important contributor to the probability of significant noncore melt releases, and that steam generator tube degradation was a major contributor to occupational radiation exposure at pressurized water reactors. Based upon these analyses, the NRC developed recommended actions in the following areas:

- Prevention and detection of loose parts and foreign objects
- Steam generator tube inservice inspection
- Secondary water chemistry program
- Condenser inservice inspection program
- Primary-to-secondary leakage limits
- Reactor coolant system (RCS) iodine activity limits
- Safety injection signal reset

The licensee addressed the above recommendations, as follows:

- Procedure SO-M-10, "Tool Accountability," established tool accountability standards for primary and secondary sides of the steam generators when their access ways are open.
- The licensee performs full-length, end-to-end tube inspections during eddy current examinations of the steam generators.
- Secondary chemistry guidelines and operating limits, which are consistent with the recommendations of both Combustion Engineering and Steam Generator Owners Group II, have been adopted and implemented.
- The licensee committed to take prompt and prudent corrective action in the event that secondary chemistry operating limits, including those relating to condenser inleakage, are exceeded.
- The TS limit the primary-to-secondary leakage through the steam generators tubes to 1 gpm total for both steam generators.
- The TS limit the specific activity of the RCS for both iodine and noniodine activity.
- At the FCS, the safety injection pumps take suction from the SIRWT. The switchover of safety injection pump suction from the boric acid storage tanks to the SIRWT does not apply at the FCS.

The licensee inspected the tubes in both steam generators, using eddy-current testing techniques, during the 1988 refueling outage.

The testing indicated that no tubes required plugging. This was the second refueling outage in a row where no steam generator tubes were plugged. The NRC recently stated in NRC Inspection Report 50-285/89-19 that a principal reason for FCS not experiencing problems with steam generator tubes appeared to be due to the strict secondary water chemistry program established by the licensee. The area of secondary water chemistry controls was inspected during routine regional inspections. No problems were noted during these inspections in the area of secondary water chemistry controls. Based upon the actions taken by the licensee to address this generic letter and the results obtained, this generic letter is considered closed.

c. Generic Letter 86-04, "Policy Statement on Engineering Expertise on Shift."

This generic letter requested the licensee to submit information on the following:

° The current program for providing engineering expertise on shift.

° If the current shift technical advisor (STA) program utilizes an "equivalency" criteria to an engineering degree, describe the criteria used.

° Describe any modifications the licensee intends to propose to their current program in order to take advantage of the options identified in the Commission's Policy Statement.

The licensee's response stated that, under OPPD's program and equivalency criteria, a qualified individual would have a bachelor's degree in engineering, engineering technology, math, physics, or similar scientific discipline with appropriate training in such areas as thermodynamics and fluids.

The inspector noted that all STAs at FCS are degreed individuals and are taking an active role in day-to-day plant operations. This generic letter is considered closed.

d. Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants".

This generic letter discussed a concern that concentrated boric acid solution, or boric acid crystals formed by evaporation of water from the leaking RCS, would corrode the reactor coolant pressure boundary. The NRC requested licensees to provide assurances that a program had been implemented to ensure that boric acid corrosion does not lead to degradation of the RCS boundary.

The licensee's response included the following:

- Special Procedure SP-CSF-1, developed in response to consideration of Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," identified the principal locations of closures with carbon steel fasteners in systems containing borated water and provided for an inspection of these fasteners. In addition to bolted closures, the other locations in the RCS pressure boundary where carbon steel is found include the reactor vessel, pressurizer, and the two steam generators. Surveillance Test ST-RLT-1, "Reactor Coolant System Leak Rate Test," identifies these pressure vessels as areas to be inspected for signs of leakage during the refueling outage.

- Surveillance Procedure ST-RLT-3, "Reactor Coolant System Leak Rate Test," is performed daily to quantify the leakage from the RCS. Both known and unknown leakage is determined using this procedure.

- When leakage from the RCS is located, immediate steps are taken to correct the situation. If the leakage event appears to have safety implications, a station incident report is initiated. These reports are a vehicle for accumulating information and ensuring the proper evaluation and disposition of the event.

The inspector reviewed the results of SP-CSF-1 performed during the last refueling outage. No problems were noted. This generic letter is considered closed.

No violations or deviations were identified.

14. Assessment of the Local Public Document Room (94600)

The inspector visited the local public document room (PDR), located in downtown Omaha, Nebraska, at the W. Dale Clark Library, to assess the content and condition of the files. The PDR consisted of four bookcases located in the southwest corner of the second floor. Some of the information that was available is listed below:

- Updated Safety Analysis Report

- Emergency Plan

- NRC Inspection Reports

- NRC Information Notices and Bulletins (information prior to August 21, 1986, was indicated as being available on microfiche)

- NRC Generic Letters

- Licensee Event Reports

- NRC News Releases
- NRC Annual Reports
- General Correspondence

A review of the files indicated that the information was indexed and maintained in an up-to-date status. The overall condition of the files was excellent and the information was easily retrievable. Although the PDR was not listed in the library directory (government documents were), the inspector asked two employees as to the location of the PDR. Both were able to give adequate directions.

No violation or deviations were identified.

15. Exit Interview

The inspector met with Mr. G. R. Peterson, Plant Manager, and other members of the licensee staff on December 5, 1989. The meeting attendees are listed in paragraph 1 of this inspection report. At this meeting, the inspector summarized the scope of the inspection and the findings. During the exit meeting, the licensee did not identify any proprietary information to the inspector.