ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket No:	50-285
License No:	DPR-40
Report No:	50-285/97-15
Licensee:	Omaha Public Power District Fort Calhoun Station FC-2-4 Adm. P.O. Box 399, Hwy. 75 - North of Fort Calhoun Fort Calhoun, Nebraska
Facility:	Fort Calhoun Station
Location:	Blair, Nebraska
Dates:	June 15 through August 2, 1997
Inspectors:	W. Walker, Senior Resident Inspector V. Gaddy, Resident Inspector
Approved:	W. D. Johnson, Chief, Project Branch B
Attachment:	Supplemental Information

EXECUTIVE SUMMARY

Fort Calhoun Station
NRC Inspection Report 50-285/97-15

This routine announced inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 7-week period of resident inspection.

Operations

- The performance of the operations staff during this inspection period was good. Operators were observed operating the plant in a professional manner (Section O1.1).

- The inspectors noted that the material condition of the auxiliary feedwater and raw water systems was good. Minor discrepancies between the piping and instrumentation diagram and operating instruction were noted during the walkdown of the auxiliary feedwater system (Section O2.2).

- Verification of two tagouts found all components in their correct position (Section O2.2).

- Ineffective corrective action resulted in a work request sticker not being removed from the control room panel as required by procedure. This was a violation (Section O2.3).

Maintenance

- The inspectors observed multiple maintenance activities during the report period. Overall, the maintenance and surveillance activities were thorough and performed professionally (Section M1.1).

- Ineffective corrective action resulted in the lower disc wedge of the boric acid totalizer bypass valve being broken due to overtorquing. This was a second example of a violation (Section M8.1).

Engineering

- Engineering personnel performed a sound technical analysis that resulted in a decrease in the correlated nuclear detector well temperature (Section E1.1).

- The licensee provided adequate justification to support the cable routing deviation identified by the inspector (Section E8.1).

Plant Support

- Due to a procedural misunderstanding, the self-contained breathing apparatus regulator flow test records were not documented appropriately. Once pointed out by the inspectors, the test records were correctly documented (Section R2.1).

- While touring the radiation controlled area, the inspectors identified that locked doors could not be opened due to improper maintenance on the door locks. This item will remain unresolved (Section R2.2).

- The licensee determined that the criticality monitor in the new fuel receipt area was not sensitive enough to detect a criticality accident (Section P2.1).

- The inspectors identified that licensee personnel did not establish any compensatory measures prior to blocking the sprinkler system in the diesel generator room. This was a violation (Section F1.1).

Report Details

Summary of Plant Status

The Fort Calhoun Station began this inspection period operating at essentially 100 percent power and operated at that level throughout the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety conscious; specific events and noteworthy observations are detailed in the sections below.

O2 Operational Status of Facilities and Equipment

O2.1 Review of Equipment Tagouts (71707)

The inspectors reviewed the following tagouts:

- Serial Number 97-0607, Replace Circuit Board of 125 VDC Battery Charger 2
- Serial Number 97-0715, Isolation of East Raw Water Header

The inspectors found that all tags were on the proper components and components were in the proper tagged position.

O2.2 Engineered Safety Feature System Walkdown (71707)

The inspectors used Inspection Module 71707 to walk down portions of the auxiliary feedwater and raw water systems. The systems were walked down using the following drawings and procedures:

- Operating Instruction OI-AFW-1, Auxiliary Feedwater Actuation System Normal Operation, Revision 33
- Operating Instruction OI-RW-1, Raw Water Normal Operation, Revision 47
- Standing Order SO-O-44, Administrative Controls for the Locking of Components, Revision 53
- Drawing 11405-M-100, Raw Water Flow Diagram
- Drawing 11405-M-254, Condensate Flow Diagram

- Drawing 11405-M-253, Steam Generator and Blowdown Flow Diagram

The inspectors noted the material condition of the equipment was good. All supports and seismic restraints were properly anchored and in good condition. Valves and breakers were verified to be in the correct positions. However, during the auxiliary feedwater system walkdown, the inspectors noted a few minor discrepancies between the operating instruction and the piping and instrumentation diagram. Specifically, the inspectors noted that Valve FW-1317 (emergency feedwater storage tank condensate makeup bypass valve) was shown as being locked on the operating instruction, but was not locked on the piping and instrumentation diagram. Valves FW-333 (emergency feedwater storage tank level alarm upper isolation valve) and FW-334 (emergency feedwater storage tank level alarm lower isolation valve) were shown as closed on the operating instruction, but were locked closed on the piping and instrumentation diagram. Valves FW-661 (emergency feedwater storage tank condensate makeup inlet valve) and FW-652 (emergency feedwater storage tank demin makeup water inlet valve) were shown as closed on the operating instruction, but were shown as open on the piping and instrumentation diagram. The as-found position of the auxiliary feedwater valves was consistent with the operating instruction.

The inspectors inquired about the discrepancies. Licensee personnel stated that they relied on the operating instruction to show the current plant valve position. The licensee initiated Document Change Engineering Change Notice 97-246 to change the piping and instrumentation diagram to ensure that they match the operating instructions.

O2.3 Failure to Remove Work Request Sticker

a. Inspection Scope (71707)

The inspectors followed up on inadequate corrective action to ensure work request stickers were removed from the control room following maintenance.

b. Observations and Findings

On June 17, 1997, while walking down the control room panels, the inspectors noted Work Request Sticker 970291 attached to the motor-driven fire pump. Work request stickers were used to visually identify items requiring maintenance. The work request sticker indicated that the pump had tripped on overcurrent. The inspectors asked a licensed operator if the pump had been repaired. The licensed operator stated that he thought the work had been completed. The inspectors asked that, if the work had been completed, why was the work request sticker still attached to the control room panel. The inspectors questioned whether all operational staff were aware of the status of the equipment.

The inspectors determined that the pump had been repaired and the paperwork had been closed out on June 9, 1997. The inspectors reviewed Maintenance Work Documents 972000 and 972031. These work documents documented work that was performed on the pump. The inspectors reviewed Standing Order SO-M-101, "Maintenance Work Control," and noted that Step 5.10.8 directed maintenance personnel to remove work request stickers following the completion of postmaintenance testing and/or operability testing.

The inspectors identified a similar incident in December 1995 that is documented in NRC Inspection Report 50-285/95-24. This involved the failure to remove a work request sticker from the control room panel following maintenance on Charging Pump CH-1C. The licensee documented this incident on Condition Report 199600119. The condition report indicated that the work request sticker was not removed following maintenance and raised the concern that failing to remove the sticker following repair could potentially result in operators not knowing the status of control room equipment. The licensee performed a root cause analysis and initiated eight action items to prevent recurrence. All the action items were closed.

Corrective action from the December 1995 incident included revising Standing Order SO-M-101 to require the shift supervisor to verify work request stickers were removed from control room panels prior to closing out a maintenance work document. This verification should include seeing the removed sticker attached to the maintenance work document. This action item was closed in May 1996.

On February 11, 1997, operations personnel performed a walkdown of the control room and identified six work request stickers attached to control room panels whose work documents were either closed or voided. Earlier corrective action did not prevent recurrence. This deficiency was documented on Condition Report 199700155. Although the root cause analysis was completed in March 1996, several subsequent condition reports have documented failures to remove deficiency stickers following maintenance.

The corrective actions implemented by the licensee to ensure that work request stickers were promptly removed from the control room panels have not been effective. Failing to ensure adequate corrective action is a violation of 10 CFR Part 50, Appendix B, Criterion XVI (50-285/97015-01).

c. Conclusions

The inspectors identified that the corrective action implemented by the licensee to ensure work request stickers were removed from the control room following maintenance was ineffective.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707)

The inspectors observed all or portions of the following activities:

- Replacement of Raw Water Pump AC-10C
- Leak repair on Steam Generator Blowdown Sample Cooler SL-8
- Replacement of lube oil circulating pump for Diesel Generator 1
- Installation of water curtain in auxiliary building stairway
- Numerous maintenance activities on diesel-driven fire pump

b. Observations and Findings

The inspectors found the work performed under these activities to be professional and thorough. All work observed was performed with the work package present and in active use. On occasion, the inspectors observed system engineers monitoring progress, and quality control personnel were present whenever required by the procedures.

c. Conclusions

The maintenance activities observed were conducted in a controlled and professional manner.

M1.2 Surveillance Activities

a. Inspection Scope (61726)

The inspectors observed all or portions of the following activities:

- OP-ST-VA-0001, Monthly Hydrogen Purge System Test, Revision 6
- OP-ST-ESF-0009, Channel "A" Safety Injection, Containment Spray and Recirculation Actuation Signal Test, Revision 28
- IC-ST-RC-0001, Functional Test of Acoustic Flow Monitors, Revision 3

- CH-CP-01-6784, Calibration of Feedwater Heater Number 6 Hydrazine Analyzer, Revision 2

b. Observations and Findings

The inspectors noted that the surveillances were performed in accordance with procedures. The surveillance procedure was present and in use during the observations. Communications between personnel performing the tests and the control room operators were good.

c. Conclusions

The surveillance activities observed by the inspectors were completed in a controlled manner and in accordance with procedures.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) Inspection Followup Item 50-285/97003-03: Broken Boric Acid Totalizer Bypass Valve CH-462. Specifically, the lower disc wedge of the valve was broken. This item remained open pending the licensee's search of vendor manuals to identify specific torquing and valve wrench requirements for manual valves and the completion of a root cause analysis to determine why the valve failed. The vendor manual for this valve indicated that the valve should tightly seat with less than 60 foot-pounds of torque on the handwheel and that no more than 100 foot-pounds of torque should be applied to the handwheel. The valve vendor indicated that the valve would begin to fail when 145 foot-pounds of torque were applied, and the lower wedge of the valve would break into five pieces when 165 foot- pounds of torque were applied. Valve CH-462 was used to isolate the 2-inch recirculation line for Boric Acid Tank CH-11B. Failure of the valve to isolate caused boric acid pump flow to indicate low and the broken valve pieces had the potential to cause safety related equipment failures.

On June 20 ,1997, the licensee completed the root cause analysis to determine what caused Valve CH-462 to fail. The licensee concluded: the lower wedge of the valve failed due to overtightening in the closed direction; the failure was most likely caused by an individual using additional leverage when tightening the valve onto its closed seat; and that physical evidence indicated that Valve CH-462 was broken due to excessive force on the handwheel.

Additionally, the licensee identified that there were over 1000 manual valves with specific torquing limitations in use in various applications. The licensee determined that potential existed for other similar valve failures to occur. The root cause analysis indicated that a more restrictive valve wrench program would be implemented by September 30, 1997. The inspectors asked if any interim guidance had been provided to operators. The inspectors were informed that training had been provided to operations personnel to sensitize them to valve overtorquing issues.

In NRC Inspection Report 50-285/95-24 the inspectors addressed an incident that occurred on December 18, 1995, in which the yoke bushing of Valve CH-172 (Charging Pump CH-1C inlet isolation valve) failed while Charging Pump CH-1C was being isolated during the performance of Surveillance Procedure OP-ST-CH-3003, "Chemical & Volume Control System Pump/Check Valve Inservice Test." A similar failure occurred to the yoke bushing for Valve CH-174 (Charging Pump CH-1A inlet isolation valve) on December 22 while isolating Charging Pump CH-1A during the performance of the same procedure. The licensee initially told the inspectors they suspected the valve failures were due to lack of lubricant being applied to the valve's stem.

Condition Report 199500441 documented the failures of the yoke bushings. The condition report discussed three actions items to prevent recurrence. Action Item 2, completed May 17, 1996, requested that operations and training recommendations be provided to enhance operator training in the area of valve manipulation to minimize the potential for overtightening valves. During the training, which was provided during Training Rotation 96-02, the importance of proper valve operation was discussed and operators were told that several recent valve failures were the result of overtightening. Also discussed were ways to reduce the occurrence of valve overtightening. Operators were also encouraged to provide input to system engineering or operations management if they had ideas to reduce instances of overtightening valves.

Corrective actions in response to Condition Report 199500441 were not effective and did not prevent Valve CH-462 from being damaged by to overtightening. Licensee review of the failure of Valve CH-462 concluded that there was a need for a more restrictive valve wrench program. Failing to assure adequate corrective action is a second example of a violation of 10 CFR Part 50, Appendix B, Criterion XVI (50-285/97015-01).

M8.2 (Closed) Unresolved Item 50-285/96018-03: missing test documentation. The item remained open pending the location of documentation to support tests performed on Main Steam Line Radiation Monitor Isolation Valves HCV-921 and -922. Originally, the licensee could not produce adequate documentation to show that tests performed by the vendor on these valves were within the scope of the maintenance work document. The licensee contacted the vendor and retrieved the test documentation. The inspectors reviewed the test data and verified that tests performed by the vendor were within the scope of the maintenance work document.

III. Engineering

E1 Conduct of Engineering

E1.1 Nuclear Detector Well Cooling

a. Inspection Scope (37551)

The inspectors followed up on the circumstances surrounding a high temperature alarm on Nuclear Detector Well TIA-733B.

b. Observations and Findings

On July 20, 1997, a high temperature alarm was received on Nuclear Detector Well TIA-733B. The temperature was approximately 152°F. This represented the annulus exit temperature from the nuclear detector cooling system. The detector well concrete temperature was then determined from Figure TDB III-24 in the Technical Data Book. The figure correlated the temperature relationship between the detector well temperature and the concrete temperature. Based on the correlation, the concrete temperature was approximately 147°F. The Technical Specification limit for concrete temperature was 150°F. If this temperature was exceeded, the plant would have to be shut down in accordance with Technical Specification 2.01.

The licensee determined the primary cause of the high temperature was high river temperature. The river temperature was approximately 82°F.

Normally, the plant operated with one component cooling water pump in operation. Operations personnel started a second component cooling water pump to increase the flow to the nuclear detector well. The increased flow did not reduce the detector well temperature.

Engineering personnel performed an investigation to determine the best resolution to the cooling problem. Engineering personnel concluded that hardware fixes without extensive modification would be ineffective in gaining sufficient margin for detector well temperature and that changes to the component cooling water flow balance posed an unacceptable risk to the reactor coolant pump seals. Engineering personnel then investigated whether the correlated temperature figure in the Technical Data Book contained sufficient margin to allow the temperature limits to be adjusted upwards. During the investigation, engineering personnel used the 10 CFR 50.59 process to more clearly define the Technical Specification reference temperature as the concrete bulk temperature. This resulted in the temperature limits being increased 4.8°F. Engineering personnel then performed another 10 CFR 50.59 applicability screening to revise the detector well temperature versus concrete temperature figure in the Technical Data Book.

With the adjusted temperature limits, the new correlated concrete temperature was approximately 144°F.

c. Conclusions

The inspectors concluded that the actions taken by engineering personnel to revise the detector well and concrete temperature correlation figure were technically sound.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved item 50-285/96018-05: potential cable routing deficiencies. This item remained open pending the inspectors' review of an additional analysis provided by engineering personnel. The analysis was intended to show that the cable routing deviation noted by the inspectors was acceptable and that the analysis justified the deviation. The inspectors reviewed the analysis and determined that the analysis appeared to justify that the deviation was acceptable.

IV. Plant Support

R2 Status of Radiation Protection and Control Facilities and Equipment

R2.1 Respiratory Equipment (71750)

During this inspection period, the inspectors performed an inspection of the respiratory equipment maintenance program. The maintenance and inspection frequency for respiratory equipment was controlled by Procedure RP-507, "Inspection and Maintenance of Respiratory Equipment." The inspectors noted that the self-contained breathing apparatus regulator flow test records were not completed for 1996. In response to the inspectors' findings, the licensee performed an investigation and determined that the test had been completed but, due to a procedural misunderstanding, the results had been incorrectly documented by radiation protection personnel. In response, radiation protection personnel were counseled and the procedure was changed to clearly state what tests were required to be documented. No other anomalies were noted.

R2.2 Inspection of Areas Not Easily Accessible in Auxiliary Building

a. Inspection Scope (71750)

On July 11, 1997, the inspectors selected several rooms in the radiation controlled area to inspect. These areas were either designated as contaminated or restricted high radiation areas. The rooms designated as restricted high radiation were locked, as was one of the rooms which was used for storage.

b. Observations and Findings

The inspectors observed the radiation protection technician attempt to unlock Shutdown Heat Exchanger Valve Room 15A. The radiation protection technician was unable to unlock the door. A review of maintenance history on this door and other doors in the auxiliary building was conducted by maintenance personnel. This review determined that eight doors which had been serviced utilizing Procedure GM-RM-FP-AO1, "Fire Door Lockset Inspection and Repetitive Maintenance," Revision 5, on July 3, 1997, had the tumbler assembled backwards. This did not affect doors which were normally open in that the doors would still open. However, any door which was locked with the tumblers reversed could not be opened using the keys, without disassembling the locking mechanism. Maintenance personnel identified that the door to the volume control tank room, Room 29, was also locked and would not open.

The licensee determined that it took approximately 30 minutes to open the doors with the appropriate tools.

The inspectors questioned the licensee concerning the impact on operations of these doors being locked and not easily accessible. The licensee reviewed the components in these rooms and determined that both rooms contained equipment that was referenced in the abnormal operating procedures and/or emergency operating procedures. The licensee concluded that, in an accident condition, not having access to the equipment in the rooms for approximately 30 minutes would not impact plant safety. The licensee also performed a reportability evaluation and determined that the condition was not reportable. This item will remain unresolved pending the inspectors' review of the assumptions in the reportability evaluation and review of the abnormal and emergency operating procedures (50-285/97015-02).

c. Conclusions

During a tour of the radiation controlled area, the inspectors identified that locked doors could not be opened because the tumblers had been incorrectly installed. This item will remain unresolved pending the inspectors' review of the abnormal and emergency operating procedures.

P2.1 Criticality Accident Requirements

a. Inspection Scope (92904)

The inspectors assessed the licensee's compliance with 10 CFR 70.24. The inspectors used guidance entitled "Inspection Plan for Compliance with 10 CFR 70.24, 'Criticality Accident Requirements,' at Operating Nuclear Plants."

The inspectors interviewed several licensee personnel and reviewed the following documents:

- Updated Safety Analysis Report Section 11.2.3.8, "Area Radiation Monitors";
- Updated Safety Analysis Report Section 9.5.3.3, "New Fuel Storage";
- Condition Report 199600382, "Fuel Handling";
- Condition Report 199600990, "Fuel Handling Equipment Operation";
- Condition Report 199700796, "New Fuel Storage Rack";
- Operating Instruction OI-FH-1, "Fuel Handling Equipment Operation";
- Area Radiation Monitor Alarm Setpoint Validation RP-216;
- Maintenance Procedure RE-RI-FE-0701, "Receipt of New Fuel";
- Operations Technical Specification Required Shift Surveillance OP-ST-SHIFT-0001;
- Surveillance Test IC-ST-RM-012 "Calibration of New Fuel Uncrating Entrance Area Radiation Monitor RM-80";
- Surveillance Test IC-ST-RM-0001, "Quarterly Functional Test of Area Radiation Monitors";
- Operating Procedure OP-12, "Fueling Operations"; and
- Technical Data Book, Chemistry TDB-IV.8, "Area Monitoring Setpoints."

b. Observations and Findings

The inspectors examined the facility operating licensee. The licensee had not been granted an exemption from 10 CFR 70.24 pursuant to 10 CFR 70.24(d).

The inspectors reviewed the Updated Safety Analysis Report. The Updated Safety Analysis Report provided information regarding area radiation monitors. The inspectors identified no specific commitments pertaining to criticality accidents.

The inspectors reviewed the accidental criticality monitoring and alarm system. The system used gamma-sensitive radiation detectors which provided a distinct audible and visual alarm. The criticality/area radiation monitors and locations monitored were as follows:

- RM-80 - New Fuel Storage Rack/Receipt Area
- RM-87 - Spent Fuel Pool Area

The inspectors reviewed the calibration and functional test procedures for the criticality/area radiation monitors. The monitors generated a warning/alert alarm at 5 mRem per hour for RM-80 and 20 mRem per hour for RM-87.

The radiation monitors for the spent fuel pool and new fuel receipt area were addressed in Technical Specification, Section 3.1, Table 3-3, Item 3. The licensee performed a channel check of the monitors on a daily basis and a channel functional test quarterly. Channel calibrations were performed during each refueling outage.

The inspectors reviewed the licensee's practices regarding evacuations and drills required by 10 CFR 70.24(a)(3). The licensee had maintained and implemented emergency procedures for evacuation and drills. The licensee had performed drills in accordance with the emergency response plan and the emergency plan implementing procedures.

Control room operators were designated with the responsibility for determining the cause of criticality alarms. The alarm response procedure directed the control room operators to contact radiation protection personnel to investigate the alarm and enter Abnormal Operating Procedure AOP-09, "High Radioactivity."

All personnel with unescorted access receive General Employee Training, which teaches the employee to evacuate an area should an area radiation monitor activate and sound and to report it to radiation protection personnel.

Initially, the licensee informed the inspectors that both criticality monitors were capable of detecting a criticality accident. Subsequently, the licensee performed additional calculations and determined that, if a criticality accident occurred in the new fuel storage area, the actual dose rate at Radiation Monitor RM-80 would be approximately 1.55E-2 mRem per hour. That would not be sufficiently high to generate an alarm of Radiation Monitor RM-80. Based on the calculation, the licensee was considering applying for an exemption from 10 CFR 70.24. A criticality accident in the spent fuel pool could be detected by Radiation Monitor RM-87. This item will remain open to allow the inspectors to review the licensee's resolution of this issue (50-285/97015-03).

c. <u>Conclusions</u>

The licensee determined that an accidental criticality in the new fuel receipt area could not be detected by the area criticality monitor. The licensee was evaluating whether to pursue an exemption from this requirement.

F1.1 Fire Protection Impairment Requirements

a. Inspection Scope (71750)

The inspectors performed a review of the compensatory measures performed by the licensee during painting activities in Diesel Generator Room 2.

b. Observations and Findings

On July 9, 1997, the inspectors toured Diesel Generator Room 2. The inspectors noted that painting activities were ongoing and a scaffold had been erected to facilitate painting the ceiling and walls. The scaffold decking was blocking the sprinkler heads used for mitigating a fire in the diesel generator room.

The inspectors questioned the fire protection system engineer concerning compensatory measures for the diesel generator room. It was determined that a fire protection impairment permit should have been obtained and implemented prior to installing the decking to ensure that appropriate compensatory measures were in place.

The inspectors reviewed Standing Order SO-G-58, "Control of Fire Protection System Impairments." Step 5.1.4 requires that a fire impairment permit and appropriate compensatory measures be in place prior to affecting the operability of the fire suppression system.

Technical Specification 5.8.1 requires that written procedures be established, implemented, and maintained covering activities recommended in Regulatory Guide 1.33. Failing to obtain the necessary fire protection impairment permit and failing to establish a compensatory fire watch prior to installing the scaffold decking is a violation of Technical Specification 5.8.1 (50-285/97015-04).

c. Conclusions

The inspectors identified a violation of the fire protection procedures that required a fire protection impairment permit and compensatory fire protection measures prior to affecting the operability of the fire suppression system in the diesel generator room. The fire protection system engineer promptly initiated the proper impairment permit and implementation of an hourly fire watch as a compensatory measure.

VI. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on August 5, 1997. The licensee acknowledged the findings presented. During the course of the exit meeting, one issue concerning criticality monitoring was classified as an inspection followup item. The licensee informed the inspectors that a request for exemption from criticality monitoring, as required under 10 CFR 70.24, was in final review and would be submitted to the NRC in the near future. The inspectors will close the followup item after licensee submittal and NRC approval of the exemption request.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT
SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

G. Bishop, Assistant Plant Manager
J. Chase, Manager, Fort Calhoun Station
H. Faulhaber, Manager, Maintenance
S. Gambhir, Division Manager, Production Engineering
J. Gasper, Manager, Nuclear Projects
B. Hansher, Supervisor, Station Licensing
J. Herman, Manager, Outage Management
R. Jaworski, Manager, Design Engineering, Nuclear
R. Phelps, Manager, Station Engineering
R. Ridenoure, Supervisor, Operations
B. Shubert, Manager, Chemical Services

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 61726: Surveillance Observations
IP 62707: Maintenance Observations
IP 71707: Plant Observations
IP 71750: Plant Support Activities
IP 92903: Followup - Engineering
IP 92904: Followup - Maintenance

ITEMS OPENED AND CLOSED

Opened

50-285/97015-01	VIO	inadequate corrective actions (Sections O2.3 and M8.1)
50-285/97015-02	URI	inadequate maintenance on locked doors (Section R2.2)
50-285/97015-03	IFI	criticality monitors (Section P2.1)
50-285/97015-04	VIO	failure to complete fire impairment permit and establish compensatory fire watch in diesel generator room (Section F1.1)

Closed

50-285/96018-05	URI	potential cable routing deficiencies (Section E8.1)
50-285/96018-03	URI	missing test documentation (Section M8.2)
50-285/97003-03	IFI	broken boric acid totalizer bypass valve (Section M8.1)