APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-285/92-14 Operating License: DPR-40

Docket: 50-285

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

Facility Name: Fort Calhoun Station

Inspection At: Blair, Nebraska

Inspection Conducted: June 7 through July 18, 1992

Inspectors: R. Mullikin, Senior Resident Inspector
R. Azua, Resident Inspector

Approved: P. H. Harrell, Chief, Project Section C, Division of Reactor Projects — Date 8/12/92

Inspection Summary

Inspection Conducted June 7 through July 18, 1992 (Report 50-285/92-14)

Areas Inspected: Routine, unannounced inspection of review of previously identified inspection findings, licensee event report followup, onsite followup of events, operational safety verification, maintenance and surveillance observations, and safety-related system walkdown.

Results:

- Following a reactor trip on July 3, 1992, licensed operator performance in identifying plant conditions, stabilizing the plant, and performing a controlled cooldown was very good (paragraph ~~ a).
- Radiological protection personnel efforts in support of plant activities following the loss-of-coolant event were very good (paragraph 6.c).
- In the areas of radiological protection, security, and operations, management oversight of personnel activities continues to be a strength (paragraph 6).

- In response to NRC Bulletin 91-01, the licensee determined that the Fort Calhoun Station did not have any Thermo-Lag fire barrier insulation installed in the plant (paragraph 6.f).
- Maintenance activities were found to be well coordinated with good communications noted between field personnel and control room operators (paragraph 7).
- Preplanning and attention to detail by maintenance personnel was found to be a strength (paragraph 7).
- Walkdown of the auxiliary feedwater system identified the system to be properly aligned and operable (paragraph 9).

DETAILS

1. Persons Contacted

*R. Andrews, Division Manager, Nuclear Services
J. Bobba, Supervisor, Maintenance
J. Chase, Assistant Manager, Fort Calhoun Station
*R. Clemens, Supervisor, Outage Projects
*G. Cook, Supervisor, Station Licensing
*D. Eid, Engineer, Station Licensing
M. Frans, Supervisor, Systems Engineering
*S. Gambhir, Division Manager, Production Engineering
*J. Gasper, Manager, Training
*W. Gates, Division Manager, Nuclear Operations
*R. Jaworski, Manager, Station Engineering
*W. Jones, Senior Vice-President
*L. Kusek, Manager, Nuclear Safety Review Group
*W. Orr, Manager, Quality Assurance and Quality Control
*T. Patterson, Manager, Fort Calhoun Station
A. Richard, Assistant Manager, Fort Calhoun Station
J. Sefick, Manager, Security Services
C. Simmons, Station Licensing Engineer
F. Smith, Supervisor, Chemistry
*R. Short, Manager, Nuclear Licensing and Industry Affairs
J. Tills, Supervisor, Operations

The inspectors also contacted additional personnel during this inspection period.

*Denotes attendance at the exit meeting on July 23, 1992.

2. Plant Status

The Fort Calhoun Station operated essentially at 100 percent power until July 3, 1992, when a reactor trip occurred (see paragraph 5.a). The licensee experienced a loss-of-coolant event, which resulted in the declaration of an ALERT. The Fort Calhoun Station remained shut down throughout the remainder of this inspection period to perform repairs on a pressurizer code safety valve and modifications to the nonsafety-related inverters and the turbine electrohydraulic control system.

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

a. (Closed) Open Item 285/9010-01: Accuracy of the Valve Positions Indicated on Plant Piping and Instrumentation Diagrams

This item identified that the valve positions shown on piping and instrumentation diagrams may not be accurate. The concern was that the control room operators might use the piping and instrumentation diagrams, during an event, to determine the required position of valves that do not have control room position indication.

In response to this concern, the licensee revised Drawing 11405-MECH-1, "Symbol List Piping and Instrumentation Diagram," to include the note, "All valve symbols reflect actual location; however, they may not depict actual valve position for normal (Mode 1) operation. Consult applicable operating instruction for actual valve position during any mode of operation."

In addition, operators have been instructed, during simulator training, to use the operating instruction valve lists, instead of the piping and instrumentation diagrams, to determine required valve positions for valves that have no position indication in the control room. The operating instruction valve lists provide the required valve positions.

The inspector assessed the licensee's corrective actions by interviewing control room operators. The operators were found to be cognizant of this concern and indicated that only the operating instruction valve lists would be used to determine required valve positions. The interviews confirmed that the licensee has adequately addressed this item.

b. (Closed) Apparent Violation 285/9126-01: Inadequate Procedural Controls for Nonroutine Chemistry Sampling Activities

During an enforcement conference, conducted on December 19, 1991, NRC reviewed the facts associated with this apparent violation and concluded that this item was not a violation of regulatory requirements.

c. (Closed) Violation 285/9126-02: Normally Locked Containment Isolation Valve Operated Contrary to Approved Procedures

This violation resulted from the use of a valve (WD-1060), on a test line between two automatic containment isolation valves (HCV-500A and -500B), for sampling the reactor coolant drain tank. Valve WD-1060 was opened on about 20 occasions to take liquid samples from the reactor coolant drain tank discharge line. The sampling was performed contrary to procedural requirements for locked valves.

The licensee determined that the root cause of this event was a lack of formality in the sampling process. Valve WD-1060 was used as a sample point without a formal review by all the departments involved. In addition, the lack of knowledge of the sampling effort by various personnel prohibited their ability to identify the resulting containment integrity problem. Contributing causes, identified by the licensee, included no approved procedure for the nonroutine sampling activity, lack of understanding/training related to opening seal-wired closed valves, and no labeling of the seal wires on valves.

The licensee's corrective actions included:

- The establishment of management expectations for the need and implementation of formalized plans for significant nonroutine activities and proper coordination and implementation of troubleshooting or other minor activities that may affect operations.

- Revision of the applicable chemistry procedures to ensure that procedural requirements for operation of locked valves are implemented.

- Provide training on this event and Standing Order O-44, "Administrative Controls for Locking of Components," to personnel that may operate or direct operation of station valves.

- Install a label on locked valves to identify the purpose of the locks.

The inspector reviewed the licensee's corrective actions and found that the actions adequately addressed this violation. The inspectors noted, during tours of the plant, that the locked valves observed had a visible label identifying that the shift supervisor's approval was required prior to manipulating the valve.

d. (Closed) Apparent Violation 285/9126-03: Containment Integrity Not Maintained

During an enforcement conference, conducted on December 19, 1991, NRC reviewed the facts associated with this apparent violation and concluded that this item was not a violation of regulatory requirements.

4. <u>Licensee Event Report Followup (92700)</u>

a. (Closed) Licensee Event Report 91-017: Potential for Radiological Release Through the Safety Injection and Refueling Water Tank Vents

As a result of evaluations prompted by similar industry events, the licensee identified a potential radiological release path through the safety injection and refueling water tank vents. Due to possible back leakage of containment sump water to the safety injection and refueling water tank following an accident, the design basis loss-of-coolant accident dose consequences could exceed 10 CFR Part 100 and Standard Review Plan 6.4 limits. This was reported as a condition outside the design basis for the plant.

The licensee's corrective actions included:

- Performing an interim dose calculation for the identified limiting leakage to support continued plant operation.

- Replacing Valves LCV-383-1 and -2 with more effective isolation valves during the 1992 refueling outage.

- Establishing a leak test program by developing System Engineering Procedure SE-EQT-SI-0001, "Measurement of Post RAS Leakage to the Safety Injection Refueling Water Tank," to determine and monitor potential radioactive leakage paths to the safety injection and refueling water tank during a recirculation actuation signal.

- Incorporating the leak testing requirements for these valves into the inservice inspection program by September 26, 1993.

- Evaluating the radiological impact of the initial baseline leak rate test results, obtained during the 1992 refueling outage, to establish appropriate leakage acceptance criteria for the inservice inspection program.

The inspector reviewed documentation for the completion of the corrective actions. The implementation of leakage testing of the affected valves into the inservice inspection program is expected to be completed by September 26, 1993. The other corrective actions have been completed. Based upon the completed actions and the above commitment, this licensee event report is closed.

b. (Closed) Licensee Event Report 91-021: Inadvertent Containment Isolation Actuation Signal

This licensee event report documented an inadvertent partial actuation of the containment isolation actuation signal, which occurred on October 4, 1991, while the plant was heating up from Mode 4 (cold shutdown) to Mode 3 (hot shutdown). During the performance of Surveillance Procedure OP-ST-ESF-0009, "Channel A Safety Injection, Containment Spray and Recirculation Actuation Signal Test," Channel A inadvertently actuated.

The licensee's review identified that the containment isolation actuation signal override test switch did not make proper contact when operated. The contacts, which initiate the alarm indications (annunciators and amber lights), engaged before the containment isolation actuation signal blocking contacts engaged. Following an inspection of the containment isolation actuation signal override switch, it was identified that there was no defects with the switch. The licensee concluded that the improper switch contact operation was caused by not placing the test switch against its hard stop in the test position. The licensed operator trainee who had positioned the switch was not aware of this need.

As a corrective action, the licensee added caution statements to the surveillance test procedure to require that the switch be turned until it reaches its end stop. Prior to the incorporation of the procedure change, the licensee placed a caution tag on the switch to alert the operators of the potential problem with switch operation.

The inspector reviewed documentation for the completion of the corrective actions. As a result of the completed actions, this licensee event report is closed.

c. (Closed) Licensee Event Report 91-022: Nuclear Instrumentation Channels B and D Outside Design Basis

This licensee event report addressed the determination that neutron flux monitoring Channels NE-002(B) and NE-004(D), selected to meet the requirements of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," did not meet the single failure criteria. This condition was outside the design basis for the plant.

The licensee determined that the root cause of this event was the failure to follow Procedure PED-QP-5, Revision 0, "Engineering Analysis Preparation, Review, and Approval," when updating Engineering Study ES-84-07, Revision 2, following a modification effort involving neutron flux monitoring Channels B and D on November 30, 1989. This effort was in response to a 10 CFR Part 21 report regarding manufacturing defects in the neutron flux detector/cable assembly. Procedure PED-QP-5 was intended to ensure that design documents were properly updated, including reviews for technical accuracy.

The licensee's corrective actions included issuing a memorandum, to production engineering division personnel, that provided guidance on the applicability of Procedure PED-QP-5 to analysis/studies predating the August 7, 1989, issuance of the procedure, and issuing Nonconformance Report 91-096. The disposition of the nonconformance report was to bring either Channel A or C into compliance with the environmental qualification requirements of Regulatory Guide 1.97. As a result, repairs were made to the containment portion of Channel A wide-range nuclear instrumentation under Maintenance Work Order 914005 during the most recent outage.

The inspector reviewed the documentation for the completion of the corrective actions. As a result of the completed actions, this licensee event report is closed.

d. (Closed) Licensee Event Report 91-024: Unplanned Engineered Safety Feature Actuation After Pulling Fuses

This licensee event report documented an unplanned engineered safety feature actuation when an electrician, while removing fuses from inside a control panel in the control room, pulled a fuse from the wrong fuse block. The removal of the fuse caused ventilation isolation actuation signal Relays B/94-1, -2 and -3 to deenergize. Loss of power to the ventilation isolation actuation signal relays caused two containment radiation monitor sample valves to close and the control room air conditioning unit and Radiation Monitor RM-065 to start.

The licensee identified that the cause of this event was attributed to personnel error. The electrician failed to check the danger tag sheet to ensure that he was pulling the fuse from the correct fuse block.

The licensee's corrective actions included briefing electrical maintenance personnel on the lessons learned from this event, revising Standing Order M-100, "Conduct of Maintenance," to require that electrical maintenance personnel review the appropriate drawings prior to pulling fuses or lifting leads and to provide training on the already established self-checking programs.

The inspector reviewed documentation for the completion of the corrective actions. Based upon the completed corrective actions and the above commitment to develop a training program, this licensee event report is closed.

e. (Closed) Licensee Event Report 91-025: Safety Injection Pipe Supports Outside Design Basis

This licensee event report addressed the determination that the upset and faulted loadings on two safety injection system pipe supports (SIH-14 and -64) exceeded the design capacity of the embedded strut to which they were attached. This was a condition outside the design basis of the plant.

The licensee identified that the primary cause of this event was attributed to a design analysis deficiency (i.e., inadequate consideration of zero period acceleration loadings in an analysis performed by a consultant). A contributing factor was the lack of experienced licensee personnel to review the consultant's work. The licensee has since developed in-house expertise in the area of seismic analysis and has purchased a computer program that accounts for zero period acceleration.

The subject supports were inspected and showed no visual signs of unacceptability. The supports met the interim operability criteria, which was established during the 1991 refueling outage and described in a letter to the NRC, dated May 11, 1990. As a corrective action, the licensee modified the piping supports, during the 1992 refueling outage, so they met the design basis requirements.

The inspector reviewed Modification Request FC-87-14 and verified its completion. Based on the review performed by the inspector, this licensee event report is closed.

5. Onsite Followup of Events (93702)

a. Loss-of-Coolant Event and Declaration of an ALERT

On July 3, 1992, at 11:52 p.m., the licensee declared an ALERT due to a reactor coolant system leak that exceeded 40 gallons per minute. The leak occurred when Pressurizer Code Safety Valve RC-142 failed to fully reseat after lifting, which resulted in a loss-of-coolant event. This event required the operators to shut down and cool down the plant using the natural circulation mode of operation. The details of the event are documented in NRC Inspection Report 50-285/92-18, which provides the results of the event review performed by an Augmented Inspection Team.

On July 4, Region IV issued a Confirmatory Action Letter to the licensee. This letter specified the actions that the licensee was required to take to return the plant to power operation following the event. The actions included inspection and testing of Valve RC-142, inspection of the short-term corrective actions taken by the licensee, and a meeting between NRC and licensee personnel in Region IV regarding the results of the licensee's investigation of the event.

The licensee removed Valve RC-142 from the plant and shipped it to the Wyle Laboratory for inspection and testing. These activities were witnessed by Region IV personnel. A description of the results of the inspection and testing are provided in NRC Inspection Report 50-285/92-18. As noted in the inspection report, the licensee was in the process of designing and installing a locking device to capture the adjusting bolt nut and the adjusting bolt. This action was being taken to ensure that, once the safety valve had lifted, a setpoint change would not occur because the adjusting bolt could not back out. The licensee designed and installed a locking device in Valves RC-141 and -142 prior to reassembly of the valves. Both valves were subsequently tested and found to operate satisfactorily.

On July 16 a meeting was held in Region IV with NRC and licensee personnel. At the meeting, the licensee presented the short-term actions, referred to as the Recovery Plan, that were in progress to correct the equipment and system deficiencies that were identified during the event. The Recovery Plan provided by the licensee at the meeting is attached to a letter from the NRC to the Omaha Public Power District, dated July 29, 1992. During the meeting, NRC personnel reviewed the actions proposed to be completed by the licensee and noted that the list of items was satisfactory.

Subsequent to the meeting, in a letter dated July 22, the licensee provided confirmation that all items included in the Recovery Plan had been completed and that the plant was safe for restart. The inspectors performed an independent review of selected items in the Recovery Plan to verify that the licensee had taken the appropriate actions to correct the identified anomaly. A discussion of each item reviewed by the inspectors is provided below:

- Investigate Equipment Damage Inside Containment and Inspect Mechanical Systems for Effects of the Event

  Extensive tours of containment were performed by licensee and NRC personnel to identify any equipment anomalies that may have occurred as a result of the event. As a result of the detailed walkdown of piping systems, it was noted that two pipe supports (RCS-63 and RCH-42), located on the discharge piping of the power-operated relief valves, were damaged. The anchor bolts for Support RCS-63 had pulled away from the wall approximately 1/2 inch and the rod for Support RCH-42 was slightly bent.

  The licensee inspected the rod on Support RCH-42 and noted that the very slight bend did not affect the capability of the support to perform its intended safety function; therefore, no repairs were made.

  To repair Support RCS-63, the licensee installed larger anchor bolts to increase the safety factor from 2 to 4. The licensee also performed a visual inspection of all the support welds and noted no problems.

The inspector reviewed the actions taken by the licensee and noted that the appropriate documentation had been provided to instruct personnel on how to replace the anchor bolts. The inspectors verified that the anchor bolts had been replaced and the repairs were performed in accordance with the instructions provided.

- Prepare an Action Plan/Procedure to Dewater the Containment Sump

  The inspector verified that the licensee had pumped the water out of the containment sump. The water was transferred to the waste system for processing.

- Inspection of the Insulation on the Lower Portion of the Reactor Vessel

  The licensee performed a visual inspection of the lower portion of the reactor vessel and the insulation that covers it. This was done to verify that the insulation and vessel were not adversely affected when wetted by the water in the containment sump.

  The inspector reviewed the results of the licensee's inspection, including photographs taken of the lower portion of the vessel. Based on the review of the licensee's documentation, it appeared that the vessel and insulation were not adversely affected by the wetting.

- Modification of the Electrical Distribution System for Testing of Inverter 2

  As noted in NRC Inspection Report 50-285/92-18, issued by the Augmented Inspection Team, Inverter 2 could not be tested following maintenance without connecting the inverter to its normal loads. To address this vulnerability, the licensee modified the electrical system configuration to allow testing of the inverter without connecting it to its normal loads.

  The inspector reviewed the modification performed by the licensee to verify that the modification would allow testing of the inverter using an alternate source of loads. The inspector noted that the licensee had completed the modification for Inverters 1 and 2. No problems were noted during review of this item.

- Modification of the Controls for the Electrohydraulic Control System

  As identified by the review performed by the Augmented Inspection Team, a contributing factor to this event was that the turbine control valves went shut and created a loss-of-load condition because a reactor trip is not initiated when the control valves shut. As a result of the loss-of-load condition, a reactor trip occurred on high reactor coolant system pressure.

  To address this contributing factor, the licensee performed a modification that will result in a reactor trip when the control valves shut. By installing the modification, a loss-of-load condition caused by the control valves shutting will not occur.

  The inspector reviewed the documentation associated with the installation and testing of the modification. The inspector noted that the modification received the proper approvals prior to installation and the testing which was performed was appropriate. No problems were identified during the reviews.

- Replacement of the Rupture Disk on the Pressurizer Quench Tank

  The licensee replaced the rupture disk on the quench tank. The disk ruptured when the quench tank filled with coolant during the loss-of-coolant event.

  The inspector reviewed the documentation used by the licensee to replace the disk. The inspector noted that the documentation provided satisfatory instructions and the disk replacement was performed by qualified individuals. In addition, the inspector noted that the licensee performed a pressure test, after replacement of the disk, to verify leak tightness of the quench tank. Based on the reviews performed by theinspector, no problems were noted with the replacement of the rupture disk.

b. <u>10 CFR Part 50, Appendix R, Concern Due to Undersized Cabling</u>

On July 2, 1992, the licensee reported that the 4160-volt electrical cables for Heater Drain Pumps FW-5A, -5B, and -5C were undersized for fault current capability. It was determined that a three-phase fault on the feeder cables could cause the conductor temperature to exceed the jacket ignition temperature along the length of the cables. A fire in the turbine building or in the air compressor room could cause the heater drain pump cables to ignite and effect both safe shutdown trains in the switchgear rooms. This was discovered during the licensee's design basis reconstitution effort.

The licensee initiated a 1-hour firewatch in the affected areas and prepared Safety Analysis for Operability Report 92-01 to justify continued operations. The licensee based continued operation on the compensatory measures (firewatches) and on an analysis of the cable jacket and insulation design, which showed that the extent of fault damage was limited.

The inspectors will perform further review of this event during routine review of Licensee Event Report 92-022.

Conclusions

The corrective actions taken by the licensee to address the equipment anomalies identified during the loss-of-coolant event was very good. The degree of management oversight in ensuring that all items were appropriately addressed was excellent.

6. Operational Safety Verification (71707)

a. Routine Control Room Observations

The inspectors observed operational activities throughout this inspection period to verify that proper control room staffing and control room professionalism were maintained. Shift turnover meetings were conducted in a manner that provided for proper communication of plant status from one shift to the other. Discussions with operators indicated that they were aware of plant and equipment status and reasons for lit annunciators. The inspectors observed that Technical Specification limiting conditions for operation were properly documented and tracked.

Following a reactor trip on July 3, 1992, operator performance in identifying plant conditions, stabilizing the plant, reducing reactor coolant system leakage, and performing a controlled cooldown was found to be very good. Their demeanor was professional, their efforts were timely and well thought out, and communications between operations personnel during the event were also very good.

b. Plant Tours

The inspectors toured various areas of the plant to verify that proper housekeeping was being maintained. Various valve positions were verified for the correct plant conditions. Personnel were observed obeying rules for escorts and visitors and entry and exits into and out of vital areas.

On July 7, 1992, following the removal of the reactor coolant system fluid from the containment sump, the inspectors toured the containment building with licensee management personnel. All accessible locations were inspected. No deficiencies were identified by the inspectors.

c. Radiological Protection Program Observations

The inspectors verified that selected activities of the licensee's radiological protection program were properly implemented. Radiation and contaminated areas were properly posted and controlled. Health physics personnel were observed routinely touring the controlled areas. In addition, management personnel were observed touring the auxiliary building, reviewing radiation protection activities.

Following the loss-of-coolant event on July 3, 1992, radiation protection personnel support of operational and maintenance activities was found to be very good. These activities included the initial entries into the containment to establish entry requirements based on leakage from the reactor coolant system; inspection of the reactor vessel bottom; and removal of Pressurizer Code Safety Valves RC-141 and -142. Radiation protection personnel decisions were found to be conservative with forethought towards personnel safety. As-low-as-reasonably-achievable briefings were held prior to each activity to minimize personnel exposure. In addition, due to the high temperatures in containment, plant safety personnel required monitoring of blood pressure for personnel entering the containment and limited the amount of time personnel were allowed to stay in containment. Finally, good use of respiratory equipment by all personnel involved was observed.

d. Security Program Observations

The inspectors observed security personnel perform their duties of vehicle, personnel, and package search. Vehicles were properly authorized and controlled or escorted within the protected area. Temporarily designated vehicles and designated vehicles parked in the protected area were found to be locked and the keys were located with the associated departments responsible for their use, as required by security program procedures.

On June 5, 1992, the inspector monitored portions of licensee surveillance testing on the perimeter detection equipment and noted no problems. These efforts were performed in accordance with program procedures. Security management personnel have been observed touring the protected area to monitor security personnel activities.

e. Observation of Management Activities

Throughout this inspection period, management involvement in operational activities continued to be visible. Management personnel have been present during operator turnover briefings and have addressed operators during these briefings, especially following the July 3 automatic trip and subsequent loss-of-coolant event. Management personnel, located in the technical support center during the event, performed their duties in a professional manner.

f. NRC Bulletin 92-01

On June 24, 1992, the inspector delivered a copy of Bulletin 92-01 to the licensee. The bulletin addressed the potential failure of Thermo-Lag 330 fire barrier systems to maintain electrical cabling, in wide cable trays and small conduits, free from fire damage. The bulletin requested that licensees immediately identify areas with Thermo-Lag 330 and take compensatory measures. The licensee confirmed that the Fort Calhoun Station did not have any Thermo-Lag installed at the plant and that no compensatory measures were required. The licensee planned to provide, within 30 days, written confirmation of this information.

Conclusions

Operator performance following the July 3 automatic reactor trip and subsequent loss-of-coolant event was very good. Radiation protection personnel efforts in support of plant activities during this event were found to be very good.

In the area of security, personnel were found to be knowledgeable of their responsibilities. In addition, management oversight of personnel activities in the area of security, operations, and radiation protection continued to be considered a strength.

7. Maintenance Observations (62703)

The inspectors observed selected station maintenance activities on safety-related systems and components.

a. Removal of the Pressurizer Code Safety Valves

On July 7, 1992, the inspectors witnessed the removal of Pressurizer Code Safety Valves RC-141 and -142. The work was performed under Maintenance Work Orders 922896 for RC-141 and

922889 for RC-142 using Procedure PE-RR-RC-0400, "Pressurizer Safety Valve Removal." Although Valve RC-142 was the relief valve that leaked during the event, the licensee removed both valves for inspection and testing. The results of the inspection and testing are documented in NRC Inspection Report 50-285/92-18.

Due to the temperature in containment and the need for all personnel to wear anticontamination clothing (including respirators), the licensee's safety officer restricted stay time to approximately 1 hour and 15 minutes. Thus, the licensee initially planned to have two crews perform the work with the second crew entering containment as the first crew exited. The first crew was scheduled to remove the insulation off of both valves, take flange gap measurements, determine as-found bolt torque values on the inlet and outlet flanges, and remove Valve RC-141. The two inspectors divided their observations among both crews.

However, during the first containment entry, the stay time expired before the valve removal was started. The second crew removed Valve RC-141 during its allotted stay time and the first crew was required to reenter to remove Valve RC-142. The licensee's medical technician monitored each person's blood pressure before allowing reentry.

The inspectors noted that the valve removals were performed according to procedure and that good care was taken to not damage the valve upon removal. The valves were supported by a hoist, and after removal from the pipe, the valves had to be moved to the room entrance by carefully moving the valve to another hoist and then releasing the first hoist. This process was carefully performed with no apparent rush to complete before the stay time expired. Damage during removal could have hindered subsequent inspection and testing. It was noted during removal that no material came out of Valve RC-142. However, a piece of foreign material (later determined to be duct tape) came out of Valve RC-141. The valves were removed from containment and packed for shipping to the Wyle Laboratory in Huntsville, Alabama, for inspection. NRC review of the safety valve condition is documented in NRC Inspection Report 50-285/92-18.

b. <u>Raw Water Outage to Replace Valve HCV-2881B</u>

On July 14, 1992, the inspectors witnessed the replacement of Valve HCV-2881B. This valve is the raw water discharge valve for component cooling water Heat Exchanger AC-1B. Valve HCV-2881B, along with the raw water inlet valve on each of the four heat exchangers, was replaced during the recent refueling outage. However, shortly after the end of the outage, it was discovered that Valve HCV-2881B could not be opened. To remove

Valve HCV-2881B, the entire raw wate: system must be removed from service. During the plant shutdown, the licensee decided that a raw water syste outage to allow valve removal was feasible.

During the raw water system outage, the licensee was acutely aware that the component cooling water and reactor coolant systems would begin to heat up and establish limits on the maximum temperatures for the systems. The licensee determined that the component cooling water system was limiting and that it would take the component cooling water system 2 1/2 hours to reach its maximum allowable temperature. Outage time was limited to 90 minutes.

The licensee performed this effort using Maintenance Work Order 922699. Communication coordinators were set up in the control room and locally near the Heat Exchanger AC-1B. Throughout this effort good communications were maintained. The effort took approximately 21 minutes between the time that the raw water pumps were shut off and when they were restarted. The initial component cooling water temperature was 79°F and the final temperature was about 86°F. The initial reactor coolant system temperature was 100°F and the final temperature was about 105°F.

The inspectors reviewed the maintenance work order and its associated procedure and found it to be prop[illegible] approved.

Conclusion

The licensee's maintenance activities were found to be well coordinated with good communications observed between field personnel and control room operators. Preplanning and attention to detail were strengths.

8. Surveillance Observation (61726)

The inspector reviewed the licensee's local leak rate test results that were obtained during the recent refueling outage.

Procedures IC-ST-CONT-3001 and IC-ST-CONT-3002, "Type C Leak Rate Test," were found to contain the associated regulatory requirements and commitments. The tests were found to be conducted per the procedures and in accordance with Appendix J to 10 CFR Part 50 and ANSI N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors."

Throughout this test, procedural compliance was apparent. Test equipment used was found to be in calibration at the time of use, as verified by the inspector through review of calibration documentation. Testing personnel were found to have the proper qualifications to perform this effort. The completed procedure had been properly reviewed by the licensee.

Test results indicated that two valves were found t^ ·xceed their as-found leak rate limits, one of wh·^h was repai, ·, ^tested, and found to have an as-left leakage ra·. of 0 standa· · · · c centimeters per minute. The other valve is presently schedule· · · repair once spare parts are obtained. The overall as-found and ·.-left, total leakage rates were well below the 0.6La limit of approximately 62,500 standard cubic centimeters per minute.

Conclusion

Documentation of local leak rate test results was good. The licensee's review of the test results was good and deficiencies were appropriately addressed.

9. Safety-Related System Walkdown (71710)

The inspector walked down accessible portions of the auxiliary feedwater system and verified the correct valve and switch positions. The valve locations and system configuration were verified using the appropriate piping and instrumentation drawings. The valve positions indicated in these drawings were further verified by comparing them to Procedure OI-AFW-1, "Auxiliary Feedwater Actuation System Normal Operation." No errors or discrepancies were observed.

The overall condition of the system piping and valves was good. No valve packing leaks or other notable valve damage, such as bent valve stems, missing handwheels, or improper labeling, was identified. The emergency feedwater storage tank was found to contain the proper volume of water, as required by Technical Specifications.

Conclusion

The inspector concluded, based on verification of system status, that the auxiliary feedwater system was capable of performing its intended safety function.

10. Summary of Open Items

The following is a synopsis of the status of all open items generated and closed in this inspection report.

Licensee Event Reports 91-017, 91-021, 91-022, 91-024, and 91-025 were closed.

Apparent Violations 285/9126-01 and -03 were closed.

Violation 285/9126-02 was closed.

Open Item 285/9010-01 was closed.

11. Exit Meeting

The inspectors met with Messrs. W. C. Jones (Senior Vice-President) and W. G. Gates (Division Manager, Nuclear Operations) and other members of the licensee staff on July 23, 1992. The meeting attendees are listed in paragraph 1 of this inspection report. At this meeting, the inspectors summarized the scope of the inspection and the findings. During the exit meeting, the licensee did not identify as proprietary, any information provided to, or reviewed by, the inspectors.