APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-445/93-26
50-446/93-26
Operating Licenses: NPF-87
NPF-89

Licensee: TU Electric
Skyway Tower
400 North Olive Street, L.B. 81
Dallas, Texas 75201

Facility Name: Comanche Peak Steam Electric Station, Units 1 and 2

Inspection At: Glen Rose, Texas

Inspection Conducted: May 30 through July 10, 1993

Inspectors: D. N. Graves, Senior Resident Inspector
W. B. Jones, Senior Resident Inspector
G. E. Werner, Resident Inspector
T. Reis, Project Engineer

Approved: LA Yandell July 29, 1993
L. A. Yandell, Chief, Project Section B Date

Inspection Summary

Areas Inspected (Units 1 and 2): Routine, unannounced inspection of onsite followup of events, operational safety verification, maintenance and surveillance observations, followup on corrective actions for violations, and other followup.

Results (Unit 1 and 2):

- Licensee response to the indication of high reactor coolant pump stator temperature was excellent (Section 2.1).
- Contamination control during the Unit 2 surveillance outage was excellent (Section 3.2).
- System status control errors continue to pose a concern and one example of a poor operating practice regarding component position control was identified (Sections 3.3 and 3.4).
- Auxiliary feedwater (AFW) system check valve backleakage continued to distract operators during the Unit 2 startup (Section 3.5).

- Troubleshooting activities associated with a Unit 2 feedwater isolation valve were well conducted and controlled (Section 4.5).

- Extensive system engineering involvement and guidance was observed during the various troubleshooting activities (Section 4.10).

- Although most maintenance and surveillance activities were well conducted, one violation was identified regarding a Unit 2 motor-driven AFW pump maintenance activity (Section 5.2).

Summary of Inspection Findings:

- Violation 446/9326-01 was opened (Section 5.2).
- Violation 445/9247-02 was closed (Section 6.1).
- Inspection Followup Item 446/9311-01 was closed (Section 7.1).

Attachment:

- Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

At the beginning of this inspection period, Unit 1 was operating at full power. The unit remained at full power until a manual reactor trip was initiated on June 26 in response to an indicated high stator temperature on Reactor Coolant Pump (RCP) 1-04. The unit was restarted on June 27 and returned to full power where it operated for the remainder of the inspection period.

Unit 2 was in Mode 5 conducting a surveillance outage at the beginning of this inspection period. The outage was completed on June 26 with the reactor startup conducted on July 3. Mode 1 was entered on July 7 and the unit was at approximately 70 percent power with the licensee preparing to reenter the initial startup test program at the end of this inspection period.

2 ONSITE RESPONSE TO EVENTS (93702)

2.1 Manual Reactor Trip of Unit 1

On June 26, 1993, at 1:36 a.m., the Unit 1 reactor was manually tripped due to an indicated high stator temperature of 319°F on RCP 1-04.

The event began at 12:18 a.m., on June 26 when a computer generated alarm was received on RCP 1-04 indicating a stator temperature of 297°F. Abnormal Operating Procedure ABN-101, "Reactor Coolant Pump Trip/Malfunction," Revision 4, was referenced. No specific separate procedure existed for a high RCP stator temperature. Shift management was notified and the Instrumentation and Control (I&C) department was requested to verify the temperature indication. The temperature element was disconnected from the computer and directly measured by the I&C staff which confirmed that the detector output was the same as the computer indication. Five additional temperature detectors mounted in the stator were not accessible outside the biological shield which prevented them from being used for confirmation purposes. Operations personnel compared pump current and component cooling water parameters for RCP 1-04 to the three remaining RCPs and determined that the current was approximately 30 amperes higher and that the cooling water return temperature was several degrees warmer, although still within procedural limits and not increasing.

Several actions were performed by the shift operators to reduce the indicated rising temperature on the RCP 1-04 stator including increasing cooling water flow to the motor, and raising bus voltage to the pump to reduce motor amperage. Indicated stator temperature continued to increase. At 1:36 a.m., following reconnection of the detector to the plant computer and observing a stator temperature of 319°F, the shift supervisor ordered the containment evacuated, the reactor manually tripped, and the subsequent tripping of RCP 1-04. The emergency operating procedures for a reactor trip were entered and the plant was stabilized in Mode 3. All safety-related equipment functioned as required.

Unit 2 was in Mode 5 at the time of the trip with the startup of secondary systems in progress. As a result of the trip, a lockout relay on the west offsite bus actuated causing a loss of station service Transformer 2ST which was supplying the nonsafety buses on Unit 2. This resulted in a loss of power to the circulating water, condenser vacuum, turbine plant cooling water, and electro-hydraulic control oil systems. No safety-related systems on Unit 2 were affected. The lockout relay actuation was determined to be the result of a failed sersor which was subsequently replaced. The west bus was restored, and Transformer 2ST and all Unit 2 nonsafety buses were reenergized at 3:09 a.m., on June 26.

The NRC Operations Center was notified of the reactor trip in accordance with the requirements of 10 CFR 50.72(b)(2).

Subsequent troubleshooting by I&C personnel concluded that the temperature element in the RCP 1-04 stator had failed, but had failed gradually, resulting in the false indication of increasing stator temperature. Winding and resistance tests on the motor determined that no motor malfunction existed. Although six temperature elements were mounted in the RCP motor stator, only one was connected to provide indication of stator temperature. The remaining temperature elements were tested and the computer was reconnected to a functional temperature sensor. The reactor was restarted on June 27, with power restored to 100 percent on June 28.

Additional actions taken by the licensee included a change to Procedure ABN-101 to provide additional guidance regarding confirmation that an actual high temperature condition exists. The licensee was also reviewing the feasibility of a design modification to provide additional channels of stator temperature indication that would be accessible outside the containment biological shield. Reviews of design data were also conducted by the licensee to identify other indications or parameters that provided little or no redundancy and whose associated procedures mandated a manual reactor trip based on those indications.

The inspectors reviewed the licensee's response to the event, including the post-trip review conducted in accordance with Operations Department Administrative Procedure ODA-108, "Post RPS/ESF Actuation Evaluation," Revision 4, and no deficiencies were identified. Additional evaluation will be conducted in conjunction with the review of the licensee event report associated with this event.

2.2 Unit 2 Reactor Coolant System Unidentified Leakage

At approximately 1:15 a.m., on July 2, with the Unit 2 reactor in Mode 3, the licensee performed an initial Unit 2 reactor coolant system water inventory in accordance with Procedure OPT-303, "Reactor Coolant System Water Inventory," Revision 5. The results indicated that the unidentified leakage was approximately 1.2 gpm, which was greater than the limit of 1.0 gpm allowed by

Technical Specification 3.4.5.2b. The associated action requirements for the identified condition stated that the leakage must be reduced to within the allowable value within 4 hours, or the reactor be placed in Mode 3 within the next 6 hours, and Mode 5 within the following 30 hours.

The licensee entered Abnormal Operating Procedure ABN-103, "Excessive Reactor Coolant System Leakage," Revision 3, and began investigating possible leakage paths. Two additional leak rate calculations were performed and both indicated approximately 1.1 gpm. No elevated radiation levels, sump levels, or increased humidity were detected inside containment.

Licensee management decided that upon expiration of the 4 hours allowed by the Technical Specifications a Notification of Unusual Event (NOUE) would be declared in accordance with Emergency Plan Procedure EPP-201, "Assessment of Emergency Action Levels, Emergency Classification and Plan Activation," Revision 8. At 5:15 a.m., the source of the leakage had not been determined and the NOUE was declared. The shift supervisor assumed the role of emergency coordinator and directed the appropriate initial notifications of local and state agencies and NRC.

Subsequent to the notifications, the licensee determined that a drain valve from the RCP seal injection Filter 2-02, 2CS-8386B, was leaking a solid stream of water approximately 1/8-inch in diameter. The isolation valves for the seal injection filter were tightened and the observed leak rate decreased to a rate of several drops per minute.

At approximately 9:15 a.m., a reactor coolant system water inventory was completed and an unidentified leak rate of approximately 0.7 gpm was calculated. However, a review of the calculation by the shift supervisor determined that the change in reactor coolant system temperature between the initial and final sets of data was outside the procedural tolerance and concluded that the results of the test were inconclusive. A new leak rate determination was initiated.

During the performance of the second water inventory, the inspectors observed that the level in Safety Injection Accumulator 2-01 was at the upper end of its allowable level and pressure band. A review of the readings recorded at approximately 8:30 a.m., on the control room logs, indicated that the accumulator level had increased approximately 5 percent. The leak rate calculation was completed and indicated a reactor coolant system unidentified leak rate of 0.17 gpm. System engineering determined that the increase in accumulator level due to reactor coolant system leakage would result in an unidentified leak rate of 0.15 gpm. The excess accumulator level was drained and pressure and level in the accumulator were restored to normal, midrange levels. Restoring the accumulator level resulted in seating of the check valve between the accumulator and the reactor coolant system since no further increase in accumulator level occurred.

Based on the results of the leak rate calculation, the NOUE was terminated at 1:05 p.m., on July 2.

2.3 Conclusions

Initial operator response to the addressed events was excellent. Notifications were initiated as required. Immediate corrective actions were appropriate. Management involvement was evident in the determination of operator actions in both events, and a conservative operational philosophy was demonstrated during the RCP high stator temperature event.

3 OPERATIONAL SAFETY VERIFICATION (71707, 92701)

3.1 Plant Tours

General plant tours were conducted by the inspectors to assess the material condition of installed equipment, general plant cleanliness, control of combustibles, and industrial safety.

General housekeeping throughout the plant was determined to be very good. Temporary storage of material such as ladders and anticontamination clothing was appropriate.

Numerous small leaks were observed inside the radiologically controlled area on both contaminated and noncontaminated systems. All observed leaks had been previously identified and the leakage was contained with catch basins and routed to floor drains. One significant leak was observed in the Unit 2 steam generator blowdown penetration room. The area had been enclosed to prevent the leak from presenting a personnel hazard. At the end of this inspection period, the licensee had scheduled repairs for the leaks as soon as plant conditions allowed the steam generator blowdown system to be taken out of service.

Transient combustibles were observed to be kept to a minimum and none were observed to be improperly stored.

Tours inside the Unit 2 containment indicated that the areas were being maintained free of nonrequired material, and work areas were appropriately roped off. No significant leaks were identified by the inspectors inside the Unit 2 containment.

3.2 Radiation Protection Observations

Contamination control inside the Unit 2 containment during the surveillance outage was observed to be excellent. General area access required no special clothing or protection. System leakage was properly contained and routed to floor drains.

Radiation and contaminated area survey maps were found to be current and posted in easily visible locations near the areas and rooms surveyed.

3.3 Control of Feedwater Isolation Valve Position

On June 30, 1993, while performing a Unit 2 control board walkdown following testing of the feedwater isolation valves, the inspectors observed that all four feedwater isolation valves were open. A review of the unit log by the inspectors did not identify any log entries that would indicate the valves were open or the reason for being open. The reactor operator, when asked by the inspectors, did not know why the valves were open. The inspectors then queried the unit supervisor who indicated that the valves were open so that the nitrogen accumulators could be recharged with the valves in the open position. The inspectors reviewed the paperwork associated with the testing of the valves, Test Procedure OPT-511B, "FW Section XI Isolation Valves," Revision 1. The documentation indicated that the testing was completed and the independent verification step of the restoration section, which required the valves to be shut, had been completed even though previous steps of the restoration section had not been performed. A review of watch station turnover sheets in the control room by the inspectors determined that the unit supervisor's sheet contained a note stating that the feedwater isolation valves and bypasses should be shut following recharging of the nitrogen cylinders. Although the operations department administrative procedures allow procedure steps to be performed out-of-sequence with unit/shift supervisor approval, and the entry on the unit supervisor's turnover sheet satisfies the intent of the administrative procedures regarding control of components manipulated outside of prescriptive procedures, the lack of a unit log entry indicating the manipulation of major components and the lack of awareness by the reactor operator of the valves' positions and purpose was identified by the inspectors to the licensee as a poor operating practice.

3.4 System Status Control Errors

Through attendance at the daily plan of the day meetings and review of operations notification and evaluation (ONE) forms, the inspectors observed that additional examples of system status control problems had been identified. While none of the examples caused any operational problems or directly impacted the ability of any safety system to carry out its intended function, the continued occurrences caused the inspectors to question the effectiveness of previous actions taken by the licensee to address this concern and to review any additional actions taken to preclude additional instances.

The operations department initiated several actions to minimize the potential for system status control problems. Recommendations or suggestions from the operating staff by operations management regarding configuration/system status control were formally solicited. The use of a valve manipulation log sheet was implemented to record the manipulation of any valve performed without specific procedural control. The log was to record the component operated, the time, the reason manipulated, and the as-left condition of the component.

The log was to be reviewed on a shiftly basis by the field support supervisor. A task team of operations personnel made up of volunteers will review each new occurrence of system status control errors and compare it to previous events for generic or common concerns. The team was intended to include auxiliary operators, radwaste operators, reactor operators, and unit supervisors.

3.5 Unit 2 Power Escalation

On July 7, 1993, the inspectors observed Unit 2 licensed operators' activities in the control room during the power escalation associated with the reactor startup. The reactor was steady at approximately 15 percent power. Preparations were being made to prepare Steam Generator Feedwater Isolation Valve 2HV-2134 for postmaintenance testing (Refer to Section 4.5 for a description of the postwork test). The unit supervisor provided excellent control and coordination of maintenance, testing, and operational evolutions. Good use of procedures and self-checking were demonstrated by all control board operators.

The inspectors did note that the unit supervisor and balance-of-plant operator were routinely distracted from normal evolutions by the repeated AFW check valve backleakage. The operators were required to start the AFW motor and turbine driven pumps a total of four times in attempts to cool the AFW lines and reseat the check valves. Although the check valve backleakage was not safety significant, the required monitoring and pump starting activity routinely distracted the operators from other control board monitoring activities.

3.6 Conclusions

General housekeeping was very good. Although a number of leaks existed in the plant, they were generally well contained and documented. Radiation protection activities and contamination control during Unit 1 operations and the Unit 2 surveillance outage were excellent.

Although minor instances of system status control problems continued to occur, the implementation of additional corrective action initiatives is a positive indication of licensee management's continuing efforts to improve configuration/system status control and reduce personnel errors.

4 MAINTENANCE OBSERVATIONS (62703)

4.1 Motor-Driven Auxiliary Feedwater Pump (MDAFWP) 2-01 Packing Adjustment

Mechanical maintenance technicians were observed adjusting the packing on MDAFWP 2-01 (Work Order 1-93-045032-00). The pump had been recently repacked and the packing was being adjusted during a 30-hour pump run in an attempt to compress the packing, thereby allowing the gland follower to be adjusted inside the stuffing box. After adjusting the packing, the technicians measured the trueness of the gland follower and recorded surface temperature readings on the gland follower with a contact pyrometer. These work practices ensured the packing was not overtightened and subsequently prevented packing damage due to overheating.

The inspectors also reviewed ONE Form 93-1191 dated June 6, 1993, which documented problems with MDAFWP 1-01 pump packing. During the performance of the monthly pump operability test (OPT-206A), the auxiliary operator noted steam coming from both inboard and outboard pump packing glands. The pump was immediately secured. The pump packing was adjusted later that same day and no further problems with overtightened packing were observed. The inspectors found that no operability concerns were identified on the ONE form and Technical Evaluation 93-1273. The maintenance group had identified the possibility of improper packing "run-in" as the cause for the above condition. Additional guidance was requested by mechanical maintenance from engineering concerning acceptable packing gland leakoff and proper packing "run-in."

4.2 Motor Control Center Breaker Cleaning

The inspectors observed electrical maintenance technicians performing cleaning, inspection, and testing on Breaker 2EB3-1/6J/Bkr, "U2 SFGD Loop A Component Cooling Wtr Return Valve 4512 Motor Breaker," using Work Order 1-93-042403-00. This maintenance activity was being accomplished since the previously scheduled preventive maintenance activity (3-92-309166-01) had not been accomplished. ONE Form FX-92-99 documented the failure to perform the maintenance activity and Technical Evaluation 93-571 documented the acceptability of postponing the activity until the next available work window when operations could support the work.

The technicians cleaned the breaker assembly and verified that no loose or heat damaged components were present. The load side resistance was measured and found to be within tolerance. Good work practices and excellent use of procedures were observed throughout the activity.

4.3 Primary Water Valve Repair

Mechanical maintenance technicians were observed reworking Valve 2SS-18S501, "Main Generator 2-01 Stator Winding Primary Water Supply Header Isolation Valve." A Siemens engineering technical representative was present during the valve assembly process to assist in the proper assembly of the valve. The valve had recently been reworked and installed in the system; however, after initial testing yielded low primary water flow, it was discovered that the valve had the incorrect parts installed for several internal components. The mechanics utilized good work practices.

During the review of Work Order 1-93-049820-00, the inspectors noted that no torque values were given for the body-to-bonnet or flange fasteners. The work order simply stated that the fasteners should be "snug-tight." After questioning the mechanics and the Siemens representative about the lack of torque values in the work document, values were obtained and included in the work package. The tightening of the fasteners to a "snug-tight" value did not appear to meet the intent of Procedure MSM-GO-0209, Revision 1, "Torque Standard with Correction Factor for Adaptors." The mechanical maintenance department was reviewing this apparent procedural discrepancy.

4.4 Recalibrate N16/Tavg Channel

I&C technicians were observed obtaining "as-found" data for the recalibration of Amplifier Card 2-JY-0420J. Procedure INC-7712B, "ACOT and CH Calibration N16/TAVG, Loop 2, Prot. Set III, CH. 0421B," Revision 0, directed the collection of data as specified by Work Order 1-93-047750-00. The inspectors verified that the data collection instrumentation had been calibrated. Both technicians demonstrated excellent self-verification skills.

4.5 Unit 2 Feedwater Isolation Valve Troubleshooting

The inspectors observed troubleshooting and testing activities associated with Feedwater Isolation Valve 2-HV-2134 performed in accordance with Work Order 1-93-046984-00. The troubleshooting was to determine the cause of the valve failing to fully open during the surveillance outage.

The valve actuator is a pneumatic-hydraulic actuator manufactured by Borg-Warner. A motor-operated hydraulic pump provides the pressure below the actuator operating piston necessary to open the valve. A nitrogen charged pneumatic cylinder provides pressure above the operating piston to drive the valve shut. As the valve opens, and the nitrogen above the operating piston is compressed, a check valve in the pneumatic system relieves the pressure above the piston into the pneumatic cylinder. Troubleshooting activities determined that the check valve was sticking in the closed position and not allowing the pneumatic pressure above the piston to bleed back into the pneumatic cylinder. During valve opening, the pneumatic pressure above the piston would overcome the hydraulic pressure below the operating piston and the valve would stop in a mid-position. In parallel with the check valve were two solenoid valves which open on a valve close signal to provide the nitrogen pressure in the pneumatic cylinder to the top of the operating piston to close the valve. Solenoids also open in the hydraulic portion of the actuator to dump the hydraulic fluid and pressure below the operating piston back to the hydraulic sump to ensure the valve closes.

Technical Evaluation 93-1375 was generated by the licensee to address the valve's operability with the sticking pneumatic check valve. The determination was made that valve operability was not affected by the sticking check valve because it was only needed to function to open the isolation valve. Per the Design Basis Document DBD-ME-203, the safety function of the valve is closed, thus the safety function was unaffected. To open the valve, a jumper was installed around one of the two solenoid valves in the pneumatic system to allow the pressure built up above the piston to reach the pneumatic cylinder through the jumpered solenoid. The jumper was then removed and the solenoid was closed. On a close signal, the solenoids open to provide the motive force for closing the valve. The inspectors reviewed the technical evaluation and found that it addressed the issue of valve operability thoroughly.

Postmaintenance testing for the work included operability testing in accordance with test Procedure OPT-511B, "FW Section XI Isolation Valves," Revision 1. At the end of this inspection period, the valve was open and caution tagged indicating that the solenoid needed to be jumpered to open the valve following a closure.

The troubleshooting activity was well conducted. Coordination between mechanical maintenance, operations, and engineering was excellent, with the system engineer directing the troubleshooting activity in the field.

4.6 Feedwater Isolation Valve Testing

The performance and test group coordinated with operations to test Valve 2-HV-2134, "Steam Generator 2-01 Feedwater Isolation Valve." Test PPT-TP-93B0-21, "Feedwater Isolation Valve 2-HV-2134 Closure Test," Revision 0, was being conducted to confirm the operability of the valve following maintenance, and to collect data to be used in troubleshooting previous failures of the valve to close in the required time interval. The valve stroked closed in approximately 3.8 seconds which was less than the 5 seconds maximum required by Technical Specification 3.7.1.6.

Work control planning did not effectively coordinate the testing with the work group and operations. Operations was instructed during shift turnover that the test was scheduled and ready to commence; however, support personnel were not on site and had to be called. Additionally, equipment necessary to prime the valve hydraulic pump and charge the nitrogen accumulators was not staged. This lack of coordination delayed power escalation causing additional operational focus to be shifted to correcting AFW system check valve backleakage which was exacerbated by the low feedwater flows.

4.7 Radiation Monitor Pump Replacement

On June 7, 1993, the inspectors observed portions of work in progress on Design Change Notice 5508, "Replace Roots Model AF-22 Blower with Thomas Pumps on Radiation Monitors 2-RE-5502/03/66." The work was implemented per Work Order MM 93-215. The inspector observed portions of the implementation for Radiation Monitor 2-RE-5503 only.

The design change notice was found to specify changes to the Unit 2 Particulate-Iodine-Gas radiation monitor skids which reflected the mounting and installation of a new pump to replace the obsolete removed pump. The vendor had supplied a pump replacement kit which included all parts, installation instructions, design documents and equipment qualification reports.

The inspectors reviewed the licensee's work order and vendor supplied modification drawings and found the package to be somewhat cumbersome; however, proper document cross referencing was found. The inspectors interviewed the I&C craftsmen who were performing the installation. They also indicated that they had initially found the work package to be cumbersome but indicated it was comprehensive and once it was understood, it was readily implemented. The craftsmen were performing the installation for the sixth time (three pumps per unit). In summary, the craftsmen were found to be knowledgeable of their required task and actively using the work package provided by the planning department.

The inspectors reviewed the licensee's technical justification for the modification and found that the licensee had properly considered the electrical impact of the different pump design. The licensee concluded the electrical impact was bounded by current motor control center loading, feeder cable sizing, and circuit breaker sizing. The radiation monitor is an emergency load requiring power from the emergency diesel generators and the licensee's analysis concluded the total loading to the motor control center and hence the emergency diesel generator was not affected. The inspectors also reviewed the licensee's 10 CFR 50.59 evaluation associated with the modification and found it to be appropriate.

4.8 AFW Check Valve Testing

The inspectors observed a reverse flow test of AFW Check Valve 2-AF-106. The leak test was performed as a postwork test following lapping o' the valve seat performed under Work Order 1-93-046663-00. The valve had been reassembled and was being reverse flow tested prior to reinstallation of the external alignment device which had been removed during the surveillance outage as a result of a misalignment that had occurred prior to Unit 2 licensing.

This test was performed by the application of demineralized water downstream of the isolated check valve and verifying that no significant leakage is observed from an upstream vent and that demineralized water pressure does not decrease more than 2 psig. The test was unsatisfactory in that both significant leakage was observed and demineralized water pressure dropped approximately 40 psig. The valve was subsequently repositioned, retested satisfactorily, and the alignment device was welded in place. Performance of the valve during the Unit 2 startup on July 7, 1993, also indicated that the valve had been properly aligned. The testing activity was well conducted and coordinated by the system engineer.

4.9 Station Service Water Valve 1-HS-4393

On June 8, 1993, the inspectors observed the performance of Surveillance Test OPT-207A, "Service Water System Operability Verification," Revision 4. During the conduct of the test station service water Valve 1-HS-4393 was used to isolate station service water flow through the diesel generator Train A heat exchanger. The operator noted that the valve indicated midposition after positioning the main control panel control switch to close.

Troubleshooting Work Order 1-93-047908-00 was initiated to determine the cause for the midposition indication. The inspectors observed the performance of the troubleshooting work order. It was found that the valve had actually closed but that the limit switch had not actuated. This was determined by taking manual control of the valve. The auxiliary operator was able to rotate the valve handwheel only a quarter turn in the closed direction before the closed indication was received. The work order was then revised to adjust the close limit switch setting.

Excellent coordination was noted between operations, electrical maintenance, and engineering personnel in assessing valve operability. It was determined that the diesel generator, which was supported by station service water, was operable during the period the work activity was in progress. This was based on operations' ability to promptly restore flow to the heat exchanger. Engineering personnel appropriately considered the effect the work activity could have on the established settings on the motor-operated valve.

4.10 Conclusions

The observed maintenance and postmaintenance testing activities were well performed and controlled. Although the majority of the activities were well coordinated and planned, some delay was encountered during the feedwater isolation valve testing due to poor staging of equipment and personnel. Extensive system engineering involvement was evident in both the feedwater isolation valve, AFW check valve, and service water troubleshooting activities.

5 SURVEILLANCE OBSERVATIONS (61726)

5.1 Residual Heat Removal (RHR) System Testing

The inspectors observed the performance of ASME Section XI testing for Train A RHR system components. The testing was conducted in accordance with Work Order 5-93-501764-AA and Procedure OPT-203A, "Residual Heat Removal System," Revision 5, Section 8.3, "Mode 1, 2, and 3 RHR Pump Test." Valve stroke times and pump performance data were collected and determined to satisfy operability requirements. Communications and procedural adherence were observed to be excellent.

5.2 ASME Section XI Testing of MDAFWP 2-01

On June 8, 1993, the inspectors witnessed selected portions of the performance of Procedure OPT-206B, "Motor-Driven Auxiliary Feedwater Pump (MDAFWP) Testing," for MDAFWP 2-01 and the associated pump packing replacement. The inspectors found that the pump met the specified ASME Section XI differential pressure and discharge flow parameters. Vibration readings taken on the pump were also well within specifications. Immediately prior to the performance of the surveillance, mechanical maintenance had replaced the pump mechanical packing. The performance of the surveillance test provided an opportunity to operate the pump and adjust the packing. After the final adjustment of the packing, the licensee ran the pump for approximately 4 hours. Immediately prior to terminating the surveillance test, the inspectors observed the pump and found there was no thread protrusion on the inboard packing gland studs, whereas the three other MDAFWPs were found to have at least 7/8-inch of packing gland stud protrusion through the stud nuts. The inspectors were concerned that too much packing may have been installed. Mechanical maintenance was consulted and the maintenance documentation supported that the appropriate number of packing rings had been installed.

Additionally, on June 16, 1993, the inspectors identified that the inboard packing gland follower on the MDAFWP 2-01 was not installed within the stuffing box. Although Vendor Manual CP-0007-001 and Maintenance Procedure MSM-GO-7210, "Pump Packing," Revision 2, do not specifically state that the gland follower be inside the stuffing box, vendor and licensee pictorial drawings show the follower inside the stuffing box. The gland follower was designed to be located within the stuffing box to prevent packing extrusion and possible pump shaft damage caused by the follower coming into contact with the rotating pump shaft. Th stuffing box provides the external boundary which provides structural support to the pump packing.

The inspectors reviewed Work Order 1-93-045032-00 and found that the package contained Design Change Authorization (DCA)-82592, Revision 0. This DCA clarified the packing requirements for the MDAFWPs. One of the packing requirements stated that the packing gland follower had to be inside the stuffing box. During the review of the work order, the inspectors identified that the pump had been released for operation by the assistant mechanical maintenance manager with the gland follower installed outside the stuffing box. The DCA was changed without the required engineering review and justification specified by Procedure ECE 5.01-03, "Design Change Notices and Related Process Documentation," Revision 4, Section 6.1.8, "Revising DCN's." The failure to follow procedures, which allowed an unanalyzed design change of the safety-rel[illegible] MDAFWP to occur, was identified as a violation of Technical Specification 6[illegible] and Procedure ECE 5.01-03 (446/9326-01).

Attempts were made to adjust the packing so that the gland follower would be within the stuffing box. Previously, due to the Technical Specification requirements for AFW in Modes 1 through 3, the licensee had only been performing an 8-hour AFW pump run to allow for packing adjustments. However, engineering management indicated that during Mode 5 plant conditions, no restrictions or operability concerns prevented a longer pump packing run-in. The licensee removed one ring of packing from each end of MDAFWP 2-01 and performed additional run-in and adjustment of the pump packing to reduce gland leak-off. No Technical Specification requirements for AFW were violated since the reactor was in Mode 5. The inspectors have subsequently observed the packing gland leakoff and have no further concerns with the packing gland follower installation or amount of water leakoff.

The inspectors disc[illegible] the above violation with the assistant mechanical maintenance manager. [illegible] manager was not aware that the work package contained DCA 82592. [illegible] indicated that the technicians had called from the field and asked for the release of the pump with the packing follower approximately 3/32 of an inch from entering the stuffing box. Based upon his discussions with the technicians concerning pump packing performance, the assistant mechanical maintenance manager authorized the pump packing configuration. The inspectors had the following concerns with the observed work activity:

- The technicians calling from the field did not identify to the manager that the installed configuration differed from the requirements of the DCA.
- No field verification of actual equipment installation or review of work package documentation was accomplished prior to giving authorization for the installed packing configuration.
- The work supervisor who reviewed the completed work package did not identify that the as-installed configuration differed from the work package requirements even though a comment in the remarks section of the package identified the condition.

These concerns were discussed with licensee management.

5.3 Unit 1 Emergency Diesel Generator Testing

The inspectors observed surveillance testing of the Emergency Diesel Generator 1-02. The test was conducted in accordance with Procedure OPT-214A, "Diesel Generator Operability Testing," Revision 6. Communications between the control room and the diesel generator room were excellent. Synchronization and loading of the diesel generator were well performed and the operators demonstrated a cautious and deliberate attitude toward diesel generator operations. The test equipment was determined to be within current calibration dates.

The acceptance criteria contained in the OPT were verified by the inspectors to be consistent with the requirements of Technical Specification Surveillance Requirement 4.8.1.1.2a.4, -2a.5, and -2a.6. Preliminary review of the test data indicated that the results were acceptable.

5.4 Unit 2 Integrated ESF System Testing

The inspectors observed portions of integrated safety system testing performed in accordance with test Procedure PPT-P2-9900, "Reactor Trip/Engineered Safety Features Utilization," Revision 0. A test briefing was conducted by the test engineers and was attended by the control room operators as well as the shift supervisor. The briefing was thorough and included a discussion of evolutions to be performed by the test engineers, the actions required of the control room operators, and the potential risks for actuations if the procedure was not followed. Questions were asked by the operators and the test personnel responded clearly and accurately. No deficiencies were observed during the observed testing activities.

5.5 Solid State Protection System Testing

The inspectors observed the activities associated with the performance of surveillance testing on the A train of the solid state protection system.

A pretest briefing was conducted by the designated operations system expert. The briefing was attended by the unit supervisor, two reactor operators, an auxiliary operator, the system engineer, and a representative from the nuclear overview department. The briefing discussed precautions and limitations, including a recent change to the procedure, OPT-445B, "Solid State Protection System Train 'A' Actuation Logic Test," Revision 1, that installed a jumper to defeat a trip signal on the A train that was present due to the draining of Steam Generator 2-01. Unit 2 was in Mode 5 at the time. The briefing was completed and preparations were made to begin the test. The reactor operator at the controls questioned the affect of having the steam generator drained when Train B of the protection system was returned to normal from the current Mode 5/6 lineup. Further discussions between the designated system expert and the system engineer determined that the procedure change would not have prevented an ESF signal from being processed on the B train. A step in a secondary procedure referenced in the OPT, SOP-711, "Reactor Protection System," would have directed the operator to verify that no trip signals were present prior to continuing.

The test was terminated and postponed until after the steam generator was refilled at which time it was completed with no deficiencies noted. An internal TU Electric memorandum was reviewed which discussed this incident and indicated that the long term corrective action would be to modify the OPTs to jumper the steam generator low level signals in both trains of the protection system.

5.6 Unit 2 Turbine Driven AFW Pump Testing

The inspectors observed the activities associated with the surveillance testing conducted on the turbine driven AFW pump. The test was conducted in accordance with test Procedure OPT-206B, "Auxiliary Feedwater System," Revision 1. The test equipment utilized was within its current calibration date, and the steam admission valve stroke times were within the acceptance criteria. The test was temporarily delayed because the portable, strap-on flowmeter had been placed on the incorrect suction pipe. The flowmeter was placed on the correct pipe and the test was completed satisfactorily.

5.7 Containment Spray System Testing

The inspectors observed the activities associated with the testing of the Containment Spray Pump 2-02 in conjunction with protection system Slave Relay K645 testing. The tests were performed in accordance with

Procedures OPT-205B, "Containment Spray System Test," Revision 2, and OPT-477B, "Train B Safeguards Slave Relay K645 Actuation Test," Revision 1. The slave relay portion of the observed testing was performed satisfactorily, but the portion of the containment spray pump test that required measuring pump flow could not be performed because of the failure of Valve 2CT-0049, the containment spray test header isolation valve. The valve was operated via a jointed reach rod, and the fitting associated with the joint in the reach rod broke allowing the handwheel to spin freely. The test header isolation valve was verified in the closed position by the lack of indicated test header flow. A work request was initiated to repair the valve operator and the test was terminated at that point. The current surveillance on Containment Spray Pump 2-02 is valid until July 27, 1993.

5.8 Conclusions

The observed surveillance activities were generally well coordinated and conducted, although some delays were observed due to procedural or hardware deficiencies. The briefing conducted prior to the solid state protection system surveillance was thorough. The system awareness and questioning attitude of the reactor operator prior to performance of the test was commendable. Communications between the operators in the control room and in the field was considered a common strength during all the observed tests.

6 FOLLOWUP ON CORRECTIVE ACTIONS FOR VIOLATIONS (92702)

6.1 (Closed) Violation 445/9247-02: Control of Tools

This violation involved the use of carbon steel wire brushes on stainless steel components as well as the failure to properly segregate tools used on both carbon and stainless steel. The licensee performed an evaluation as to the possible effects that iron particles deposited from using improper tools would have on the stainless steel surface and found that no consequential degradation would occur. Additionally, the licensee removed all carbon steel wire brushes color coded for use on stainless steel. The inspectors concluded that the licensee's corrective actions were appropriate.

7 FOLLOWUP (92701)

7.1 (Closed) Inspection Followup Item 446/9311-01: Response to Site Evacuation Alarms

This item pertained to determining licensee management's expectations regarding actions for site personnel to take when a site evacuation alarm is sounded with no followup announcement.

The various group managers met, along with licensee senior management, and determined what their expectations were regarding personnel response. These expectations were that for alarms with no subsequent announcement, the individuals should contact their supervisor/manager for instructions or guidance. If unable to contact them, the control room should be called for clarification. Ignoring the alarm was unacceptable. This guidance was promulgated in the form of a memorandum distributed to all site personnel.

Queries of six individuals selected at random from various departments by the inspectors concluded that they were generally aware of the requirements and actions to take if no announcements were made following an alarm. No additional followup of this item was planned.

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

C. Bhatty, Site Licensing
W. G. Guldemond, Manager, Independent Safety Engineering Group
J. J. Kelley, Vice President, Nuclear Operations
D. M. McAfee, Manager, Quality Assurance
J. W. Muffett, Manager of Technical Support & Design Engineering
J. E. Thompson, Site Licensing

The personnel listed above attended the exit meeting. In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on July 9, 1993. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

bcc to DMB IE01

bcc distrib. by RIV:

J. L. Milhoan
Section Chief (DRP/B)
MIS System
RIV File
Section Chief (DRP/TSS)
W. B. Jones, DRP
Resident Inspector (2)
Lisa Shea, RM/ALF, MS: MNBB 4503
DRSS-FIPS
Project Engineer (DRP/B)
A. B. Beach, D/DRP

RIV:SRI:DRP/B	SRI:DRP/B	RI:DRP/B	PE:DRP/B	C:DRP/B	D:DRP
DNGraves:ww	WBJones	GEWerner	TReis	LAYandell	ABBeach
7/22/93	7/22/93	7/22/93	7/29/93	7/29/93	8/3/93

8/3/93