U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

NRC Inspection Report: 50-445/90-09
50-446/90-09

Dockets: 50-445
50-446

Unit 1 Operating License: NPF-28
Unit 2 Construction Permit: CPPR-127
Expires: August 1, 1992

Applicant: TU Electric
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

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

Inspection At: Comanche Peak Site, Glen Rose, Texas

Inspection Conducted: March 7 through April 3, 1990

Inspectors: W D Johnson for
S. D. Bitter, Resident Inspector, Operations — 4/19/90 Date

R M Latta
R. M. Latta, Resident Inspector, Electrical and Mechanical — 4/19/90 Date

M F Runyan
M. F. Runyan, Resident Inspector, Civil/Structural — 4/20/90 Date

W D Johnson for
R. B. Vickrey, Reactor Inspector
Region IV

4/19/90
Date

W D Johnson
W. D. Johnson, Senior Resident Inspector,
Operations

4/19/90
Date

Reviewed by: Joel S Wiebe
J. S. Wiebe, Senior Project Inspector

4/20/90
Date

Inspection Summary

Inspection Conducted: March 7 through April 3, 1990 (Report 50-445/90-09; 50-446/90-09)

Areas Inspected: Unannounced resident safety inspection including sustained control room and plant observations; operational safety verification; startup test procedure review; startup test witnessing; maintenance; surveillance testing; meeting with local public officials; followup on TMI action items (Safety Issue Management System Items II.E.1.1.1, closed; III.D.1.1.1, closed); followup of event; installation and testing of modifications; followup on previous inspection findings; followup on violations; followup on 50.55(e) reports; followup on recommendations identified by the Comanche Peak Report Review Group; followup on allegations; and Unit 2 walkdowns.

Results: Within the areas inspected, no violations or deviations were identified. Strengths included the attention to detail and degree of awareness of plant status by the operators, effective shift turnovers, effective tracking of limiting conditions for operation, housekeeping and cleanliness, and startup test coordination. A potential problem area was noted in that numerous minor leaks could lead to difficulties in contamination control.

DETAILS

1. Persons Contacted

*M. Axelrad, Newman and Holtzinger
*J. L. Barker, Manager, ISEG, TU Electric
*J. W. Beck, Vice President, Nuclear Engineering, TU Electric
*O. Bhatty, Issue Interface Coordinator, TU Electric
*M. R. Blevins, Manager of Nuclear Operations Support, TU Electric
*H. D. Bruner, Senior Vice President, TU Electric
*J. H. Buck, IAG
*R. C. Byrd, Manager, Quality Control (QC), TU Electric
*W. J. Cahill, Executive Vice President, Nuclear, TU Electric
*H. M. Carmichael, EA Manager, Unit 2
*C. B. Corbin, Licensing Engineer, TU Electric
*J. L. French, Independent Advisory Group
*W. G. Guldemond, Manager of Site Licensing, TU Electric
*T. L. Heatherly, Licensing Compliance Engineer, TU Electric
*J. C. Hicks, Licensing Compliance Manager, TU Electric
*C. B. Hogg, Chief Engineer, TU Electric
*A. Husain, Director, Reactor Engineering, TU Electric
*J. L. LaMarca, Manager of Electrical and I&C Engineering, TU Electric
*E. F. Ottney, Project Manager, CASE
*S. S. Palmer, Project Manager, TU Electric
*H. S. Phillips, Consultant, CASE
*A. B. Scott, Vice President, Nuclear Operations, TU Electric
*J. C. Smith, Plant Operations Staff, TU Electric
*P. B. Stevens, Manager of Operations Support Engineering
*J. F. Streeter, Director, Quality Assurance, TU Electric
*C. L. Terry, Manager of Projects, TU Electric
*T. G. Tyler, Director, Management Services, TU Electric
*D. A. West, Project Engineer, TU Electric
*D. R. Woodlan, Unit 2 Licensing Manager, TU Electric

The NRC inspector also interviewed other licensee employees during this inspection period.

*Denotes personnel present at the April 3, 1990, exit interview.

NRC personnel present at the April 3, 1990, exit interview:

D. D. Chamberlain, Section Chief, Region IV
T. P. Gwynn, Deputy Director, DRP, Region IV
A. T. Howell, Resident Inspector
W. D. Johnson, Senior Resident Inspector
R. M. Latta, Resident Inspector
M. L. Runyan, Resident Inspection
R. F. Warnick, Assistant Director, Comanche Peak Project Division

The following persons attended the NRC meeting with local public officials on March 27, 1990.

C. Baker, Mayor, Granbury, Texas
D. Barham, Regional Liaison Officer, Texas
S. D. Bitter, Resident Inspector, NRC
D. Cleveland, Commissioner, Hood County
G. R. Crump, County Judge, Somervell County
J. Gartrell, Mayor, Glen Rose, Texas
B. Gifford, Commissioner, Hood County
J. T. Gilliland, Public Affairs Officer, NRC
T. P. Gwynn, Deputy Director, Reactor Project Division, NRC
W. D. Johnson, Senior Resident Inspector, NRC
M. Malloy, Project Manager, NRC
M. Meyer, County Judge, Hood County
H. Strickland, City Superintendent, Glen Rose, Texas
R. F. Warnick, Assistant Director for Inspection Programs, NRC
J. S. Wiebe, Senior Project Inspector, NRC

2. Sustained Control Room and Plant Observation (71715)

In response to concerns discussed in NRC Inspection Report 50-445/90-07; 50-446/90-07, the NRC inspectors performed sustained control room and plant observation during the period March 6-11, 1990. These concerns were: (1) the adequacy of the level of attention to detail and awareness of plant status by operators, and (2) the effectiveness of shift turnover and limiting condition for operation tracking. The inspectors focused on these areas during the period of sustained control room observation.

NRC inspector findings during this period were very positive. Excellent operator attention to detail and awareness of plant status were exhibited. Shift turnovers, including control board walkdowns, were complete and performed in a highly professional manner. No errors were detected in the operating crew's tracking of limiting conditions for operation.

No violations or deviations were identified. Operator performance during the period of sustained control room observation was excellent.

3. Operational Safety Verification (71707)

The inspectors routinely toured the facility during normal and backshift hours to evaluate general plant and equipment conditions, housekeeping, and adherence to fire protection, security, and radiological control measures. Ongoing work activities were monitored to verify that they were being conducted in accordance with approved administrative and technical procedures and that proper communications with the control room staff had been established. The inspectors observed valve, instrument, and electrical equipment lineups in the field to ensure that they were consistent with system operability requirements and operating procedures.

During tours of the control room, the inspectors verified proper staffing, access control, and operator attentiveness. Adherence to procedures and limiting conditions for operations were evaluated. The inspectors examined equipment lineup and operability, instrument traces, and status of control room annunciators. Various control room logs and other available licensee documentation were reviewed. The inspectors observed and reviewed maintenance and problem investigation activities to verify compliance with regulations and procedures. Involvement of QA/QC, safety tag use, personnel qualifications, fire protection precautions, retest requirements, and reportability were considered.

Checks were made to determine whether security conditions met regulatory requirements, the physical security plan, and approved procedures. Those checks included security staffing, protected and vital area barriers, personnel identification, access control badging, and compensatory measures when required.

On April 2, 1990, Unit 1 entered Mode 2 (reactor startup). The inspector reviewed logs and records to verify that licensee personnel had completed their prerequisites for the mode change. In addition, the inspector reviewed the Technical Specification Mode 2 requirements and verified that selected requirements were met. At the end of this inspection period, Unit 1 was making an approach to initial criticality.

During plant tours, the NRC inspectors found that housekeeping was very good. A number of minor leaks were present, including flanges, fittings, vent and drain valve seat leakage, and packing leaks. Most of these had been previously identified by the licensee and were scheduled for repair. The NRC inspector noted that repair or containment would be necessary to limit contamination problems.

The NRC inspector accompanied licensee management and quality assurance personnel on a final containment walkdown inspection prior to initial criticality. While a small amount of debris was

removed during this tour, housekeeping conditions were considered to be very good. Several minor leaks, mostly associated with the chemical and volume control system were identified. The licensee initiated corrective action for the leaks.

The licensee's performance in the area of plant operations was considered to be excellent during this inspection period. No violations or deviations were identified.

4. Startup Test Procedure Review (72300, 72580)

To evaluate compliance with Regulatory Guide 1.68, Chapter 14 of the FSAR, and the Technical Specifications, the inspectors examined the test procedures listed below. The inspectors found the procedures to be thorough and technically accurate. The acceptance criteria presented in the listed procedures met the applicable requirements.

The following procedures were reviewed during this inspection period:

ISU-284A	Revision 2	Dynamic Response to Full Load Rejection and Turbine Trip

No violations or deviations were identified.

5. Startup Test Witnessing (72302)

The inspectors witnessed selected startup tests in order to verify conformance of the licensee to testing commitments and procedural requirements, observe staff performance, and to verify that adequate test program records were maintained. The following items were considered during test witnessing:

- Availability of current revision of test procedure.
- Minimum crew requirements.
- Test prerequisites and initial conditions.
- Calibration status of test equipment.
- Technical adequacy of test procedure.
- Test coordination and crew performance.
- Preliminary results satisfactory or deviations documented for further evaluation.
- Adherence to technical specifications during testing.

In addition, the inspectors reviewed various logs and reports and attended meetings and crew briefings related to the test program.

During this inspection period, the following startup tests were observed:

- ISU-021A, "Pressurizer Spray and Heater Capability."
- ISU-027A, "Hot Control Rod Operability Testing."

During these tests, coordination and communications were excellent. The procedures were adequate and no problems in crew performance were noted. No violations or deviations were identified.

6. Monthly Maintenance Observation (62703)

Station maintenance activities for the safety-related systems and components listed below were observed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, and industry codes or standards, and in conformance with the Technical Specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service, approvals were obtained prior to initiating the work, activities were accomplished using approved procedures and were inspected as applicable, functional testing and/or calibrations were performed prior to returning components or systems to service, quality control records were maintained, activities were accomplished by qualified personnel, parts and materials used were properly certified, radiological and fire prevention controls were implemented.

Work requests were reviewed to determine the status of outstanding jobs and to ensure that priority is assigned to safety-related equipment maintenance which may affect system performance.

Maintenance activities observed included:

- Troubleshooting Inverter IV1PC1 (Work Order C90-1897).
- Borg-Warner Check Valve 1MS-0196 maintenance (Work Order C90-1907, Procedure MSM-CO-8801).
- Capacitor replacement in Inverter IV1PC1 (Work Order C90-1897).

- Repair of instrument tubing leak at 1FT-0416 (Procedure INC-7762A, Work Order C90-2183).

No violations or deviations were identified.

7. Monthly Surveillance Observation (61726)

The inspector observed the Technical Specification required surveillance testing on the various components listed below and verified testing was performed in accordance with adequate procedures, test instrumentation was calibrated, limiting conditions for operation were met, removal and restoration of the affected components were accomplished, test results conformed with Technical Specification and procedure requirements, test results were reviewed by personnel other than the individual directing the test, and any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspector witnessed portions of the following surveillance test activities:

- Measurement of controlled leakage (Procedure OPT-110A).

- Turbine driven auxiliary feedwater pump test (Procedure OPT-206A). During this test, the NRC inspector observed that pump room conditions were hot and humid with a temperature of approximately 104°F. Certain steam drains were discharging steam into the room resulting in visibility being limited at times to approximately three to four feet. The system engineer was present evaluating pump performance and room conditions. Although pump operability appeared to be unaffected, licensee personnel initiated an evaluation of means to improve room conditions during pump operation. This is an open item pending completion of actions necessary to improve conditions in the turbine driven auxiliary feedwater pump room during pump operation (445/9009-O-01).

- Containment isolation valve operability test (Procedure OPT-503A).

- Cold shutdown Class 1E electrical undervoltage relay test (Procedure OPT-221A).

- Reactor coolant system wide range pressure channel calibration (Procedure INC-7727A, Work Order S90-1492).

- Cold shutdown Section XI testing of component cooling water system valves (Procedure OPT-501A).

- Fire suppression water and sprinkler system operability test (Procedure OPT-220).

- Analog channel operability test and calibration of pressurizer pressure channel (Procedure INC-772A, Work Order S89-1367).

- Primary plant ventilation system filter test (Procedure EGT-751X).

- Primary plant filtration unit testing (Procedure EGT-500-89-043).

- Train B emergency diesel generator operability test (Procedure OPT-214A).

- Backflow test of Borg-Warner check valves 1MS-142 and 1MS-143 (Procedure EGT-740A).

No violations or deviations were identified. Personnel performing surveillance tests appeared to be well qualified, and the procedures were technically and administratively adequate.

8. Meeting with Local Public Officials (94600)

The NRC personnel and the local public officials, identified in paragraph 1 of this report, met in Glen Rose, Texas, on March 27, 1990. Topics discussed during this meeting included the origin, history, and organization of the NRC; the nuclear power plant inspection program, and the transition of NRC inspection oversight of CPSES from the Office of Nuclear Reactor Regulation to Region IV.

9. TMI Action Items (SIMS)* (25565)

*The Safety Issue Management System (SIMS) tracking number is the same as the TMI Action Item number.

a. (Closed) TMI Action Item II.E.1.1.1: "Short Term Actions Concerning Auxiliary Feedwater System (AFWS) Evaluation." This item involves commitments made by the licensee in response to the NRC staff letter of March 10, 1980. NUREG-0797, "CPSES Safety Evaluation Report," (SER) documents the staff's evaluation of those commitments. Prior to the present inspection period, all but seven issues in this action item were closed by the NRC in Inspection Report 50-445/89-09; 50-446/89-09. During the present inspection period, the resident inspector reviewed these last seven issues in accordance with Temporary Instruction 2515/65. The seven issues are discussed in the paragraphs below:

- Recommendation GS-2 of the SER states (in part) that the licensee should lock open single valves or multiple valves in series in the AFWS pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFWS flow. In NRC Inspection Report 50-445/89-09; 50-446/89-09 the inspector verified that all manual valves in the AFW flow were locked open. However, the inspector noted that the manual valves in the test lines were not locked closed. He expressed concern that inadvertent, undetected opening of one of these valves could have the potential to direct a significant amount of flow away from the main flow path.

  During the present inspection period, the resident inspector verified that the AFW test discharge line manual valves are equipped with limit switches so that they annunciate at the main control board as part of the "safety system inoperable indication" (SSII) system, if opened. The resident inspector is satisfied that this is acceptable; all portions of recommendation GS-2 have been completed.

- Recommendation GS-4 stated (in part) that emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. Although this issue was addressed in NRC Inspection Report 50-445/89-09; 50-446/89-09, it was left open pending an inspector review of the procedures that address the issue of alternate AFW water supplies. During the present inspection period, the resident inspector reviewed the Emergency Response Guidelines and Procedure ABN-305A, "Auxiliary Feedwater System Malfunction." These procedures adequately address this issue. All portions of recommendation GS-4 have been completed.

- Recommendation GS-6 stated (in part) that procedures should be implemented to require an operator to determine that the AFW valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

  During the present inspection period, the resident inspector reviewed three procedures. Procedure ODA-404, "Guideline on Component Position Verification," provides the basis for establishing consistent independent verification requirements

and describes the method by which operators verify valve/breaker position. Procedure SOP-304A, "Auxiliary Feedwater System," describes the steps necessary to safely operate the auxiliary feedwater system. This procedure includes the use of attachments that list component lineups. These attachments provide a space for the qualified operator to place his initials to indicate he has verified the position of a particular component. Procedure OPT-206A, "Auxiliary Feedwater System Operability Test," describes the process by which the operators verify the operational readiness of pumps and valves in the AFW system. This procedure provides for performing independent verification in accordance with the provisions of ODA-404. The resident inspector is satisfied that these procedures adequately address the provisions of recommendation GS-6. All portions of recommendation GS-6 have been completed.

- NUREG-0797, Chapter 22, Section II.E.1.1 addressed several additional short-term recommendations that were not numbered. All but one of those recommendations were adequately addressed (and closed) in NRC Inspection Report 50-445/89-09; 50-446/89-09. The one remaining recommendation stated:

"The license should perform a 48-hour endurance test on all Engineered Safety Feature (ESF) system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 48-hour pump run, the pumps should be shut down and cooled down, and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room."

By letter dated June 16, 1981, the applicant committed to perform a 48-hour endurance test on all AFW pumps during prestartup testing. By letter dated June 24, 1981, the applicant also committed to make available test results including: (1) a brief description of the test method and instrumentation used, (2) a plot of bearing and bearing oil temperature versus time for each pump demonstrating that the temperature design limits were not exceeded, (3) a plot of pump room ambient temperature and humidity versus

time to demonstrate that the pump room ambient conditions do not exceed environmental qualification limits for safety-related equipment in the room, and (4) a statement confirming that the pump vibration limits were not exceeded.

The resident inspector has reviewed the results of the 48-hour endurance tests for all three auxiliary feed water (AFW) pumps. The test results covered the four areas described above. All portions of this recommendation have been completed.

The inspector has completed his review of this TMI action item. There are no further questions. This item is closed.

b. (Closed) TMI Action Item III.D.1.1.1: "Primary Coolant Outside Containment/Leak Isolation." This item calls for the licensee to implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. NUREG-0797 documented the results of the staff's review of the applicant's commitments that were made in the FSAR. Furthermore, Supplement 4 to NUREG-0797 documented the staff's conclusion that the licensee's limiting leakage value for liquid systems was acceptable and that the leakage value for gaseous systems would be determined by pneumatic tests.

NRC Inspection Report 50-445/89-24; 50-446/89-24 documented that this item would remain open pending the resident inspector's reviewing the licensee's leakage test program and test results. In response, the licensee submitted the procedure, STA-705, "Radioactive Systems Leakage Inspection Program," Revision 3 for review by the inspector.

The inspector has reviewed STA-705 as well as the results of the containment spray system radioactive leakage inspection test. The inspector is satisfied with these results and the program outlined in STA-705. There are no further questions. This item is closed.

10. Followup of Event (93702)

At 2:01 p.m. (CST) on March 12, 1990, an inadvertent actuation of Train A of safety injection (SI) occurred. The unit was in Mode 4 at 250°F and 380 psig. A Notification of Unusual Event (NOUE) was declared at 2:20 p.m. due to SI actuation with flow to the reactor coolant system (RCS). The NOUE was terminated at 4:25 p.m. after plant conditions were restored to normal for Mode 5. A power operated relief valve was opened for

approximately five seconds by the Cold Overpressure Mitigation System when the RCS pressure increased after a reactor coolant pump was restarted at 3:56 p.m. Train A of auxiliary feedwater started upon receipt of the SI signal, but most of the flow went to steam generator No. 1 with very little flow to steam generator No. 2.

The licensee concluded, based on tests performed on March 13, 1990, that the actuation of safety injection was caused by a shorted diode in the solid state protection system which allowed the Train A safety injection slave relays to energize when a stack radiation monitor was deenergized to replace its filter paper. The diode's purpose is to allow a safety injection signal to cause a containment ventilation isolation signal but prevent the radiation monitor from causing a safety injection signal when a high radiation signal initiates a containment ventilation isolation. Testing conducted on March 13, 1990, confirmed that the event was initiated by the failure of the diode. Portions of this testing were observed by the NRC inspectors.

The licensee promptly formed an evaluation team to review the incident.

The NRC site staff reviewed the incident including sequence of events, performance of safety equipment, cause of any equipment malfunctions, performance of operators, and effectiveness of emergency procedures.

The NRC dispatched an augmented inspection team (AIT) to the site. The site staff turned over its collected information, preliminary findings, and concerns to the AIT. The AIT inspection and findings are documented in NRC Inspection Report 50-445/90-11; 50-446/90-11.

No violations or deviations were identified. The licensee's evaluation team was promptly formed and effectively organized.

11. Installation and Testing of Modifications (37828)

The objective of this inspection activity was to observe the installation and testing of a minor design change which was implemented without being submitted to the NRC for approval. The inspection was performed to verify compliance with the requirements of Technical Specifications, 10 CFR 50.59, and 10 CFR 50, Appendix B, Criterion III.

The NRC inspector reviewed Design Modification (DM) 90-157 and associated Design Change Notices (DCNs) 835 and 837. This DM was implemented to remove the trip to automatic control function on the motor driven auxiliary feedwater pump flow control valves except when the pump gets an automatic start signal. It also changed the automatic control mode of the valves such that they

drive fully open when in automatic. After a ten second time delay, the operator can take manual control of the valves.

The 10 CFR 50.59 safety evaluation for this DM was considered to be thorough. The implementation and modification test plans were adequate. Implementation was performed in accordance with Work Order C90-2263. The NRC inspector observed portions of this installation. Testing was performed in accordance with Procedure EGT-TP-90A-11. The NRC inspector observed portions of this testing. The modified system performed as required by the test plan, meeting all of the test acceptance criteria.

No violations or deviations were identified. The design, installation, and testing of this modification indicated that the CPSES design modification process was functioning smoothly.

12. Followup on Previous Inspection Findings (92701)

a. (Closed) Open Item (445/8937-O-02): Coatings exempt log. This item was addressed in NRC Inspection Report 50-445/89-37; 50-446/89-37. It was left open pending the NRC review of the licensee's program for coating inspections and the NRC verification of its implementation.

During the present inspection period, the resident inspector reviewed Mechanical Technical Procedure EME 3.21-07, "Protective Coatings Monitoring Program," Revision 1. This procedure provides the methods and criteria for monitoring the status of protective coatings. The inspector also reviewed the results of the protective coatings preoperational monitoring walkdown performed in December 1989.

The inspector has completed his review of these items and has no concerns. This item is closed.

b. (Closed) Unresolved Item (445/8959-U-01): Questionable ability of certain safety injection isolation valves to close with a differential pressure across them. This item involved the centrifugal charging pump (CCP) safety injection isolation valves 1-8801A and 1-8801B. While the Emergency Response Guidelines (ERGs) called for these valves to be closed against a differential pressure, the licensee's response to IE Bulletin 85-03 indicated that the maximum operating differential pressure across these valves when closing was 0 psig. The inspector determined that the licensee's position needed further review since during accident conditions high radiation areas could challenge the feasibility of local operation of the valves. The licensee had committed to revise the procedures to ensure as low as reasonably achievable concerns were addressed. The inspector reviewed the licensee's changes to the ERGs. The

changes for response not obtained (RNO) were to stop the operating CCP, then close the valves and restart the CCP. Subsequent to the issue of the revised Emergency Operating Procedures (EOPs), the licensee experienced an actuation that required the use of the RNO action for valve closure. No problems were encountered during the licensee's use of the revised EOPs. This item is considered closed.

c. (Closed) Open Item (445/8959-O-04): Incorporation of appropriate resolution of inconsistencies identified by engineering into ERGs and training on these items completed before achieving Mode 2 operations. This item involved having appropriate training conducted on the revised ERGs due to a large number of changes required due to inconsistencies identified by engineering. The inspector reviewed the handout material and received a brief of the presentation for the training sessions conducted on "ERG Review Rev 5." The training sessions and material covered generic, specific step, status tree, and setpoint changes in the ERGs. The inspector was satisfied that the training had covered the incorporation of engineering inconsistency resolutions as well as other changes incorporated for Revision 5 to the ERGs. The licensee provided the inspector the attendance sheets for the above training and the inspector had no further questions of the licensee. This item is considered closed.

Licensee's action on other comments and commitments. NRC Inspection Report 50-445/89-59; 50-446/89-59 contained several comments and commitments not directly tracked as open items. These items consisted of approximately 20 issues, 50 technical comments, and 35 human factor comments. The inspector had previously followed up (see Inspection Report 50-445/90-02; 50-446/90-02, paragraph 2.g) on approximately 50% of the above issues and found the licensee to be making timely and satisfactory results with respect to their resolution. During this inspection period, the inspector followed up on the remainder of the above items. The inspector reviewed Revision 5 to the ERGs and discussed with the licensee other items that were addressed in other than ERG revisions. The inspector found that the licensee's actions with respect to its commitment to evaluate the above items and make the necessary changes was satisfactory. The licensee has taken responsible action to correct the weaknesses identified in the report.

d. (Closed) Open Item (445/8973-O-04): This item involved additional Borg-Warner check valve failures which had occurred after the initial application of corrective actions taken in response to the AFW backflow events (NRC Inspection Report 50-445/89-30; 50-446/89-30). Previous NRC inspection of this item is documented in NRC Inspection Report

50-445/90-03; 50-446/90-03. In that report, the open item was left open pending reverse flow testing of Valve 1MS-143 and the resolution of efforts to correct several check valve body-to-bonnet leakage problems. The NRC inspector witnessed the reverse flow test of Check Valves 1MS-142 and 1MS-143 per Procedure EGT-740A, Revision 1, "Reverse Flow Test for Borg-Warner Check Valves 1MS-142 and 1MS-143." Both valves tested satisfactorily and no discrepancies were observed. All Borg-Warner check valves have now been tested and verified to be operable.

An additional Borg-Warner check valve reverse flow test failure occurred during this inspection period. Valve 1AF-093 allowed excessive backflow and was observed by radiography to have the disk hung up under the top lip of the seat. The root cause of this failure was an error in the retainer height calculation. In this calculation, a gimbal of 12° around the ball bushing was assumed in lieu of the correct value of 14°. As a result of this error, the disk-arm assembly was positioned too low in the valve body, causing the same jammed-open configuration observed in this valve and others after the AFW backflow events. The retainer height calculation for Valve 1AF-093 was revised, and the valve was reinstalled and tested satisfactorily. The licensee reviewed all of the calculations in which a similar error may have occurred, and verified that the error was isolated to this specific valve. After the initial test failure and before corrective actions were taken, the test was rerun at a higher pressure and the valve seated satisfactorily. The licensee therefore concluded that the valve would have been capable of performing its design function despite the retainer elevation error. The inspector concluded that the licensee had satisfactorily addressed the identified discrepancy and generic implications.

Several Borg-Warner check valve body-to-bonnet leaks were repaired after the plant reached the normal operating temperature of 557°F. Only two check valves with minor leakage problems remain to be repaired. This open item is being closed at this time because (1) the bonnet-to-bonnet leaks do not affect the operability of the check valves and (2) the licensee has aggressively pursued this problem and remains committed to stopping the leaks. This open item is closed.

e. (Closed) Open Item (445/9003-O-05): This item involved the generic implications of a through-wall form tie hole which had been patched only at the surface and which was allowing a steady flow of air from the seismic gap into the Safeguards building. In response to NRC questions, the licensee issued Office Memorandum CPSES-9007527. In this

letter, the licensee concluded that quality measures in effect at the time the walls were constructed would not necessarily have precluded the improper surface patching of tie holes. However, many of the tie rods were left in place, limiting the number of through-wall holes which required patching. Nevertheless, for the purpose of an analysis, the licensee conservatively assumed that all form tie holes were surface patched through-wall holes. Form ties ranged in size from 3/4 to 1 1/2 inches in diameter and were spaced rectangularly on 2 and 4 foot centers. The licensee concluded that the structural integrity of existing concrete walls would not be affected even if all of the holes were unpatched. The pressure retaining function of the walls has been tested satisfactorily and will be retested periodically.

An analysis of the potential effect of surface patched through-wall holes on nearby Hilti bolts concluded that the probability of a resultant Hilti bolt failure is approximately 10E-9. This analysis did not take full credit for the fact that Hilti bolts are installed with a bolt preload that exceeds the maximum allowable design load specified by the design basis document. This suggests that any unfavorable interactions between unfilled bored holes and Hilti bolts would have resulted in rejection of the Hilti bolt upon installation.

Though abandoned Hilti bolt holes could also have been surface patched, the structural consequences are considerably less in that these holes are typically only 1 to 3 inches deep. This was the depth at which rebar was encountered, whereupon, the hole was abandoned.

The inspector reviewed Memorandum CPSES-9007527 and determined that the licensee had established a justifiable basis for closing this issue without conducting an exploratory field survey. This open item is closed.

f. (Open) Open Item (445/89200-O-05): Implementation of system status drawing program. On March 28, 1990, the turbine driven auxiliary feedwater pump was out of service for hot alignment. The safety tagout was performed under Clearance 90-0670. The NRC inspector found that the valves repositioned by this clearance had not been marked on the control room system status drawings. The unit supervisor marked the appropriate valve positions on the system status drawings upon notification. Procedure ODA-410, "System Status Control," requires components manipulated by clearances to be statused on the system status drawings. This is not a violation since the status of the system was known to the operators on duty and the system outage was logged in the limiting condition for operation tracking log.

The licensee is continuing efforts to fully implement the system status drawing program. This item remains open.

13. Followup on Violations/Deviations (92702)

- (Closed) Violation (445/9003-V-03): Inadequate review of Plant Incident Report (PIR) 89-249 for reportability. This violation involved the licensee's failure to properly implement the reporting requirements of 10 CFR Part 50.55(e) subsequent to the identification of several pieces of foreign material in the containment spray piping immediately downstream of Check Valve 1CT-145. In particular, the associated Reportability Evaluation Form SN-463 which concluded that this deficiency was not reportable failed to properly consider the implications of a potentially degraded safety-related system with regard to the establishment of the adequacy of the subject system to perform its intended function.

  TU Electric's response to this violation which was provided by letter TXX-90095 stated that since the deficiency could have rendered one train of the containment spray system inoperable, and since it is unlikely that the deficiency would have been detected by routine surveillance testing, the intent of 10 CFR 50.55(e) would have been more appropriately satisfied by conservatively reporting the deficiency. Subsequently, the inspector established that the debris in the containment spray system had been reported pursuant to 10 CFR 50.55(e) by letter TXX-90065 dated February 7, 1990, and that an inspection of the other train of the containment spray system had been performed with no foreign material identified.

  The inspector reviewed the licensee's corrective actions which included the evaluation of the cleanliness controls for fluid system internals, the examination of a sample of recently identified potentially reportable conditions which were subsequently determined to be not reportable, and the establishment of a review committee to evaluate future 10 CFR 50.55(e) reportability issues which are characterized as nonreportable. The inspector also reviewed a sample of approximately 60 recent nonconformance reports, plant incident reports, surveillances, and ONE forms which concerned system cleanliness issues. This review did not identify any additional examples of system cleanliness control deficiencies which had not been properly addressed.

  Based on the above reviews and inspection activities, it was determined that the licensee's corrective actions were adequate to rectify the identified deficiency. Therefore, this violation is closed.

14. Applicant Action on 10 CFR Part 50.55(e) Deficiencies Identified by the Applicant (37828, 51055, 92700)

a. (Closed - Unit 1 only) Construction Deficiencies (SDARs CP-89-015 and CP-89-019): "Borg-Warner Check Valve Failures." This construction deficiency involved the multiple failure of Borg-Warner check valves in the auxiliary feedwater (AFW) system which permitted backleakage from the steam generators to the condensate storage tank on two occasions (April 23 and May 5, 1989). The licensee issued SDAR CP-89-015 to address the check valve failures. An unrelated discrepancy was later discovered with a Borg-Warner check valve (1SW-048) in the service water system. In this case, the valve swing arm had broken and the disengaged valve disk was found lying in the bottom of the valve body. The licensee issued SDAR CP-89-19 to address the implications of this check valve swing arm failure. Subsequently, SDAR CP-89-19 was discontinued and both failure modes were addressed within SDAR CP-89-15.

Investigation of the four backleakage events revealed that numerous check valves in the AFW system were hung open because of an elevation difference between the disk and seat. The elevation difference, which caused the disk to become wedged under the seat lip, was created during previous reassembly activities which were based on an erroneous Operation and Maintenance (O&M) manual supplied by Borg-Warner. The O&M manual stated that the check valve retainer ring should be fully inserted to its bottom stop. However, the manual failed to state that to ensure proper disk-seat alignment, the retainer ring must be backed off to a specific elevation that is unique to each valve. As a result of implementing the erroneous instructions, the check valve disk in each case was positioned too low in the valve body to properly mate with the seat. This condition was corrected by measuring critical dimensions for each affected check valve, computing the amount of retainer backoff necessary to create proper vertical alignment, and resetting the check valve internals accordingly. The check valves were then retested by the application of hydraulic backpressure or through the use of radiography (RT). For the most part, the reverse flow tests and RT showed that the original corrective action plan had adequately resolved the discrepancy. However, several check valve test failures subsequently occurred due to rotational misalignment between the disk and seat and due to calculation errors made in the vertical alignment analysis. A special rotational alignment device was created and through its use, the alignment problems were corrected. The calculation errors were also corrected and appropriate adjustments were made to the valve internals. Eventually, all Unit 1 Borg-Warner check valves

determined to be vulnerable to the observed failure modes were retested satisfactorily.

Concurrent with the above actions, the licensee conducted an investigation of the service suitability of Borg-Warner check valve swing arms. In addition to the broken swing arm observed in Valve 1SW-048, a casting flaw was discovered in Check Valve 2CT-0148 during a maintenance inspection. The licensee retained a materials consultant, Aptech, to perform an analysis and provide recommendations. Aptech concluded that high residual stresses, a corrosive environment, and poor heat treatment during the casting process had worked together to cause the fracture in the swing arm of Valve 1SW-048. The flaw in Check Valve 2CT-0148 swing arm was attributed to inadequate casting. Aptech developed a nondestructive examination (NDE) technique to identify the swing arms which were most vulnerable to failure. This technique, which consisted of visual testing, dye penetrant testing, and replication (a metallographic etching technique for examination of metallic microstructure) was applied to all Unit 1 and common Borg-Warner check valves. Of the 80 swing arms tested, 14 failed to meet the acceptance criteria and were replaced with investment grade swing arms or swing arms borrowed from Unit 2. Eight additional swing arms were replaced for other reasons, primarily minimum wall thickness problems caused by the NDE process itself.

The NRC formed an Augmented Inspection Team (AIT) and performed an extensive review of the licensee's investigation and corrective action plan. The results of this inspection are documented in NRC Inspection Report 50-445/89-30; 50-446/89-30. Within the scope of this inspection, the AIT concluded that the applicant's corrective action plan for the check valves was adequate. Additional NRC review of issues associated with this construction deficiency is documented in NRC Inspection Reports 50-445/89-52, 50-446/89-52; 50-445/89-64, 50-446/89-64; 50-445/89-71, 50-446/89-71; 50-445/89-73, 50-446/89-73; and 50-445/89-84, 50-446/89-84. Within Inspection Report 50-445/89-73; 50-446/89-73, 12 open items were identified covering additional actions by the licensee to correct or explain apparent problem areas. All of these open items have been closed.

The licensee assessed several other technical issues closely related to or resulting from the check valve failures. These included the response of the feedwater isolation bypass valves, Check Valve 1AF-055 which had failed to fully shut during the May 5 reverse flow event, the integrity of the piping and pipe supports which had received high stresses from the intrusion of hot water, the contribution of nearby flow orifices on the failure of the AFW pump

minimum flow recirculation check valves, and the structural integrity of the affected containment penetrations. All of these issues have either been physically corrected or resolved by way of an engineering analysis.

The NRC staff concluded that the licensee has taken adequate measures to provide reasonable assurance that each Borg-Warner check valve in Unit 1 will perform its design function of preventing reverse flow. In addition, the NRC staff has found acceptable the licensee's investigation and disposition of each related technical issue listed earlier in this paragraph. Consequently, this construction deficiency is closed for Unit 1.

b. (Closed) Construction Deficiency (SDAR CP-90-03): Debris in the Containment Spray System. This SDAR involved the identification of a polishing wheel, a piece of dressing stone, and a small quantity of grinding grit in the piping downstream of Check Valve 1CT-145. As previously documented in NRC Inspection Report 50-445/90-03; 50-446/90-03, the licensee's initial evaluation of this deficiency, as described on Reportability Evaluation Form SN-463, incorrectly characterized this condition as nonreportable. Subsequent to the identification of this issue as a violation (see paragraph 13 for closeout documentation regarding 445/9003-V-03), the licensee submitted their final response by letter TXX-90065 which revised the classification of this construction deficiency and concluded that it was reportable under the provisions of 10 CFR 50.55(e).

The inspector reviewed the licensee's corrective actions which were identified in the reference TXX-90065 correspondence. These actions were previously evaluated and were determined to be acceptable during the assessment of TU Electric's response to violation 445/9003-V-03. Therefore, this construction deficiency is closed.

c. (Closed) Construction Deficiency (SDAR CP-90-04): Containment Sump Screen Holes. This SDAR concerned holes which were found in the Unit 1 inner containment sump screen. On March 9, 1990, TU Electric provided their final response to this reportable event by letter TXX-90100. As stated in this correspondence, the spacing of the fine wire mesh screen is based on the minimum size particle that could clog the Containment Spray nozzles, or grid assemblies in the reactor core, or interfere with Containment Spray or Residual Heat Removal pump clearances. Had the holes gone uncorrected, material could have entered the sump and interfered with these flow paths.

In that a definitive root cause for this deficiency could not be determined, the licensee's corrective actions were directed primarily at repairing the identified condition. These repairs were conducted in accordance with Work Order C900001635. Subsequent to these repairs, the inspector conducted a system walkdown and verified that the damaged screen had been repaired and that these activities appeared to be adequate. During the conduct of this inspection activity, several missing fasteners were identified on both No. 1 and 2 containment sump screen structures. The inspector reviewed the associated design change notices, condition reports, and ONE forms and determined that the missing fasteners had been appropriately documented and resolved.

Based on the above documentation reviews and inspection activities, it was determined that the licensee had properly characterized this construction deficiency as reportable and that the associated corrective actions were adequate to address the identified deficiency. Accordingly, this construction deficiency is closed.

15. Followup on Recommendations Identified by Comanche Peak Report Review Group (92701)

In a memorandum dated April 14, 1987, Victor Stello, the then NRC Executive Director for Operations, identified recommendations made by Guy Arlotto and other members of the Comanche Peak Report Review Group for followup. All items applicable to Unit 1 have been closed and the closures are documented in Inspection Reports 50-445/87-22, 32, and 36; 50-445/88-01, 10, 12, 15, 17, 22, 31, 40, 56, 61, and 78; 50-445/89-13; and in SSER 14.

Six items applicable to Unit 2 remain open. They are listed below and are identified by the old recommendation ID number as well as by a new open item track number:

a. Recommendation ID No. 20.

Perform a detailed review of Westinghouse's engineering evaluation regarding translation of design criteria into installation specifications, procedures, and drawings and failure to control deviations regarding Unit 2 (446/9009-O-02).

b. Recommendation ID No. 21.

Perform a visual inspection of the accessible reactor vessel surroundings during or after hot functional test (446/9009-O-03).

c. Recommendation ID No. 22.

Perform a detailed review of Westinghouse's engineering evaluation of the failure to maintain tolerance for the reactor vessel support brackets and shoes (446/9009-O-04).

d. Recommendation ID No. 46.

Evaluate traveler and NCR procedures with respect to the Unit 2 reactor vessel support brackets and shoes (446/9009-O-05).

e. Recommendation ID No. 60.

Verify the adequacy of Level I QC inspection of reactor vessel installation (446/9009-O-06).

f. Recommendation ID No. 61.

Review the completeness of QC coverage and technical acceptability of the actual inspection criteria for the installation of all the major Nuclear Steam System Supplier (NSSS) components (446-9009-O-07).

These open items will be the subject of followup inspections.

16. Allegation Followup (51053, 51055)

a. (Closed) Allegation (OSP-90-A-0011): An allegation was received by the NRC and was referred to SAFETEAM for review. The following 12 concerns were identified:

(1) While blasting rock for the construction of cooling water discharge tunnels, on two occasions, all blast materials exploded at once. The force of the explosion blew rocks the size of pickup trucks 300 feet. Although no people were injured, windows were broken up to the front gate and rock bolts were sheared off, but never replaced. These events occurred about 3 to 4 months after the start of construction.

(2) After a major concrete pour on the cooling water intake tunnel, a severe cold front arrived driving temperatures to 16°F. Although the concrete was directly exposed to the cold, the inspectors were told not to worry about this situation.

(3) Some of the holes on the inside surface of the cooling water intake and discharge tunnels were not properly grouted. Grout pumped into the holes sometimes plugged the holes before they were filled. The crew was told to go on.

(4) The final inspection of at least one tunnel was not performed because the supervisor pulled all of the lights out the night before.

(5) Some of the rebar shifted during several concrete pours. The concrete was chipped out and the rebar was repatched by just "glazing over" the bars.

(6) The alleger signed a paper stating he would say nothing detrimental about the job. He believes this was a Brown & Root document.

(7) Two concrete pumps in series were used to pour concrete, one on the outside and one inside the tunnel. Slump tests were taken on the concrete exiting the outside pump. The supervisor would not permit slump tests to be performed at the inside pump, though the slump there may have been affected by additional water being added to facilitate the flow of concrete.

(8) Rebar was supposed to touch rebar every three inches and be tied off every two ties. Instead, rebar touched every four inches and was tied off every five ties.

(9) The engineer on the cooling tunnel project had a close relationship with the project supervisor and his daughter. Consequently, the engineer may have overlooked some discrepancies.

(10) In reference to Item 1, the plant manager was told that the loud blast was a sonic boom. Rock bolts which could not be retightened were cut off and painted a beige color to blend with the rock and therefore conceal their presence.

(11) The alleger questioned the safety of the tunnel design in that the cross section was not round. The alleger felt that a circular tunnel would be safer.

(12) A large steel framing at the end of the tunnel where the cooling water rises into the plant was damaged by rock blasting, removed, and not replaced (to the alleger's knowledge). The alleger believed the framing should have been installed after all blasting work had been completed.

The NRC inspector reviewed the SAFETEAM files and compiled a summary of SAFETEAM's findings corresponding to each numbered concern:

(1) SAFETEAM interviewed several individuals who were involved with the blasting operation, but none could

corroborate an overblast of the magnitude described by the alleger. Additionally, no Class 1 structures were in place at the time of the blasting. The construction of the circulating water tunnels was appropriately nonsafety related. Photographs taken during the tunnel construction do not reveal evidence of an overblast. The rock bolts were installed for personnel safety only and were not tied to the tunnel reinforcement grid.

(2) No QC inspection was required for this nonsafety concrete placement. The concrete may have been affected if the alleger's description is accurate, but much depends on how long the cold temperatures persisted.

(3) During grouting operations, the hole was determined to be filled when the grout pump could maintain pressure for a time. It is possible that a plugged hole could have exhibited this pressure-retaining condition. However, this work was nonsafety related and a visual inspection of the tunnels in 1984 revealed that the surface concrete was adequate.

(4) This allegation could not be confirmed or denied, but this inspection would not have included any nuclear safety-related attributes.

(5) No evidence of this condition was observed during an inspection of the tunnels conducted prior to the replacement of condensers in 1987. No large patched areas or discontinuous joints were noted.

(6) A review of numerous personnel records did not reveal any such document. The alleger may have misunderstood the meaning of a form entitled "Employee Invention and Confidential Disclosure Document," which prohibits the dissemination of proprietary information. One of the blast technicians who worked alongside the alleger had no recollection of any gag order document.

(7) This configuration may have existed in certain areas of the tunnel construction but it could not be confirmed. Some concrete pumps experienced flow capacity problems that may have necessitated the use of a second pump in series. However, the insertion of water at the second pump would have been inefficient due to poor mixing and the difficulty of pumping an unhomogeneous mixture.

(8) A review of applicable drawings did not identify rebar spacing corresponding to either of the dimensions stated by the alleger.

(9) The supervisor identified by the alleger did work at the site during blasting and had a daughter employed by Brown & Root. The engineer could not be identified. Even if such a relationship existed, the work was monitored by utility engineers and other contractors.

(10) See No. 1.

(11) The tunnel design was subcontracted to a consulting firm with experience and a good reputation in the field. Photographs show that the shape of the tunnel conforms to the design.

(12) A review of tunnel drawings did not reveal the structure described by the alleger. The alleger may have been referring to arch forms which were designed to be removed.

Though several of the alleger's concerns may be accurate, the influence on plant safety appears negligible. The circulating water tunnels are not required for the safe shutdown of the reactor plant. Some indirect safety impact is implied though by the alleged gag order document (No. 6) and by the potential impact of rocks on safety-related structures (No. 1). Neither of these concerns could be substantiated. The inspector concluded that SAFETEAM's investigation of the issue was adequate. This allegation is closed.

b. (Closed) Allegation (OSP-90-A-0013): The alleger stated that his supervisor failed to provide immediate medical aid after he suffered a heart attack while at work. Additionally, a fellow worker who was splashed with cleaning solvent did not receive immediate care from the same supervisor.

The alleger submitted these concerns to the Occupational Safety and Health Administration (OSHA). OSHA determined that the supervisor was negligent in not providing immediate aid to the alleger and issued a citation and fine pursuant to 29 CFR 1926.50(a). The alleger's employer has filed an intent to contest the action. OSHA stated that an additional citation for the cleaning solvent incident is probable. Since these events are clearly within OSHA purview, this allegation is closed.

c. (Closed) Allegation (OSP-90-A-0025): Nonautomatic "F" frame circuit breakers incorrectly installed in Class 1E electrical panels. As stated in the allegation report, nonautomatic "F" frame circuit breakers were potentially installed in Class 1E electrical panels instead of the required automatic breakers during the late 1970s and early

1980s at the General Electric (G.E.) Company's Plainville, Connecticut, facility. The allegation report also indicated that these concerns were identified to G.E. management personnel; however, no response was ever provided.

In addressing this issue, the licensee performed a detailed review of the plant applications of these nonautomatic breakers which were used in both safety-related (1E) and nonsafety-related (non-1E) components at CPSES. This review indicated that a total of 14 nonautomatic 1E "F" frame circuit breakers were employed by the utility for both Unit 1, common, and Unit 2. Additionally, one nonautomatic non-1E "F" frame circuit breaker was installed per unit. As stated in the licensee's analysis of this issue, the primary difference between nonautomatic and automatic G.E. "F" frame circuit breakers was that nonautomatic breakers serve no tripping function. They are utilized for disconnecting means to transfer loads manually. Automatic breakers are auto-tripping devices. The nonautomatic breakers are utilized for manual tripping or transferring of loads at CPSES.

As determined by the inspector, nonautomatic breakers are utilized in 118, 120, 208/120V AC Class 1E distribution panel boards (CP1-ELDPPC-01, -02, -03, -04; CP1-ELDPEC-01, and -02; and CPX-ELDPEC-01 and -02). Additionally, the licensee's Calculation TNE-EE-CA-0008-183 titled "Coordination Study 118, 120, 208/120V AC Distribution Panel Boards" provides the protective device details for the Unit 1 and common panels in question.

The inspector reviewed the licensee's analysis which established that Westinghouse inverter fed panels CP1-ECDPPC-01 through -04 are protected upstream of the nonautomatic breaker with Inverter Output Breaker 4CB or the feeder breaker of Panel 1EC3 or 1EC4 when operating in the bypass mode. Also, the loads powered from CP1-ECDPP-01 through -04 have branch protection breakers which coordinate with the inverter output breaker or in bypass mode the feeder breaker of Panel 1EC3 or 1EC4.

For distribution Panels CP1-ECDPEC-01 and -02 upstream protection of the nonautomatic breakers is provided by 150A automatic breakers of Panels 1EC3 or 1EC4. Branch loads powered from CP1-ECDPEC-01 and -02 have individual breakers which coordinate with their respective device.

Unit common Panels CPX-ECDPEC-01 and -02 have protection provided upstream of the nonautomatic breakers by breakers in the 118VAC Distribution Panels 1EC5 and 1EC6 respectively. Branch loads powered from CP1-ECDPEC-01 and -02 have breakers which coordinate with their respective

upstream device. Currently the Unit 2 feeder cable to CPX-ECDPEC-01 and -02 have breakers which coordinate with their respective upstream device; however, at this time, the Unit 2 feeder cable to CPX-ECDPEC-01 and -02 are determinated. The Unit 2 Panels CP2-ECDPPC-01 through -04 and CP2-ECDPEC-01 and -02 have the same protective devices and arrangements as the associated Unit 1 panels.

Based on a review of the above documentation, it was determined that the applications of nonautomatic G.E. "F" frame circuit breakers appear to be in accordance with the previously reviewed and approved design criteria for CPSES. Therefore, this allegation could not be substantiated and is considered closed for both Units 1 and 2.

d. (Closed) Allegation (OSP-90-A-0036): It was alleged to the NRC that the reactor vessel head lifting rig was stored in the containment building in such a way that a seismic event or physical disturbance could cause it to fall into the refueling cavity. At the time of the allegation, Unit 1 was in Mode 5 with the head lifting rig stored on the reactor vessel head storage stand. The NRC inspector found that the licensee had listed the temporary storage of the head lifting rig on the head storage stand as a containment walkdown item. Safe Zone 146 had been evaluated and approved per Procedure STA-661. The lifting rig was moved to the approved safe zone on Elevation 905 prior to the plant entering Mode 3. The NRC inspector viewed the reactor vessel head lifting rig in its designated safe zone during a containment tour. This safe zone appeared to be an appropriate storage location for the lifting rig. This item is closed.

17. Unit 2 Walkdowns (71302)

During this reporting period, routine tours of the Unit 2 facility were conducted in order to access equipment conditions, security and adherence to regulatory requirements. In particular the inspector examined plant areas for fire hazards, installed instrumentation damage, system cleanliness controls, and general housekeeping. Additionally, the inspector attended several of the utility's regularly scheduled Unit 2 meetings involving project management, construction, startup, and licensing activities relative to Unit 2 schedules and programs.

As a result of these inspection activities, minor deficiencies were identified regarding delinquent preventative maintenance (PM) program issues. In particular, several instances were identified involving the failure to perform pump rotations within the prescribed preventative maintenance period and where space heater requirements were not properly implemented. Based on a

review of the pertinent maintenance records, these delinquent activities occurred prior to the assignment of PM responsibilities to Startup Engineering in February 1990.

Subsequent to Startup Engineering's assumption of this program, significant improvements have been implemented. These improvements included immediate corrective measures to reestablish the preventative maintenance program, the initiation of engineering evaluations for the affected components, and the establishment of enhanced programmatic controls. These actions appeared to be comprehensive and effective in that delinquent PMs have been reduced significantly for Unit 2 activities.

The utility's current schedule for the resumption of construction activities for unit 2 tentatively establish a April 1, 1990, date for the initiation of engineering efforts and November 1, 1990, for the start of construction. However, these dates may vary and they are provided for informational purposes only.

No violations or deviations were identified. In general, it was observed that the licensee's project management and engineering efforts which are incorporating Unit 1 lessons learned were effective. In particular, substantial emphasis has been placed on establishing a firm design package prior to the initiation of Unit 2 construction activities.

18. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Seven open items identified during the inspection are discussed in paragraphs 7 and 15.

19. Exit Meeting (30703)

An exit meeting was conducted on April 3, 1990, with the licensee's representatives identified in paragraph 1 of this report. No written material was provided to the licensee by the inspectors during this reporting period. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection. During this meeting, the NRC inspectors summarized the scope and findings of the inspection.