UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

September 13, 2000

Mr. C. L. Terry
TXU Electric
Senior Vice President & Principal Nuclear Officer
ATTN: Regulatory Affairs Department
P.O. Box 1002
Glen Rose, Texas 76043

SUBJECT: NRC'S COMANCHE PEAK STEAM ELECTRIC STATION INSPECTION REPORT NO. 50-445/00-06; 50-446/00-06

Dear Mr. Terry:

On August 19, 2000, the NRC completed an inspection at the Comanche Peak Steam Electric Station, Units 1 and 2, facility. The enclosed report presents the results of the inspection. The results of this inspection were discussed with Mr. Mike Blevins and other members of your staff.

The inspection was an examination of activities conducted under your license as they relate to safety and to compliance with the Commissions's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occurred. The violation is being treated as a noncited violation (NCV), consistent with Section VII.B.1.a of the Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or severity level of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Comanche Peak Steam Electric Station, Units 1 and 2, facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Joseph I. Tapia, Chief
Chief, Reactor Project Branch A
Division of Reactor Projects

Docket Nos.: 50-445
50-446
License Nos.: NPF-87
NPF-89

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NRC Inspection Report No.
50-445/00-06; 50-446/00-06

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-445 50-446
License Nos.:	NPF-87 NPF-89
Report No.:	50-445/00-06 50-446/00-06
Licensee:	TXU Electric
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56 Glen Rose, Texas
Dates:	July 9 through August 19, 2000
Inspectors:	A. Gody, Senior Resident Inspector S. Schwind, Resident Inspector G. Guerra, Resident Inspector W. Maier, Senior Emergency Preparedness Inspector P. Elkmann, Emergency Preparedness Inspector
Approved By:	J. I. Tapia, Branch Chief, Reactor Project Branch A

ATTACHMENTS:

Attachment 1: Supplemental Information

Attachment 2: NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

IR05000445-00-06, 05000446-00-06; on 07/09-08/19/2000; TXU Electric; Comanche Peak Steam Electric Station, Units 1 & 2; Integrated Resident & Regional Inspection Report; Mitigating Systems.

Cornerstone: Mitigating Systems

- Green. The inspectors noted that heat exchanger performance trending had not been conducted for approximately 1½ years on the Unit 1 emergency diesel generator jacket water coolers and for about 1 year on the Unit 2 emergency diesel generator jacket water coolers. During those periods, the Units 1 and 2 Train B emergency diesel generator jacket water coolers were frequently fouled beyond the acceptance criteria and were considered degraded. Failure to promptly identify this condition adverse to quality was a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation in accordance with Section VI.A of the NRC Enforcement Policy and is in the licensee's corrective action program as Smart Form SMF-2000-0001548-00 (Section 1R07).

  This issue was characterized as a green finding using the significance determination process. It was determined to have very low risk significance because the licensee's past operability review determined that the degraded emergency diesel jacket water coolers were operable.

Report Details

Summary of Plant Status

Unit 1 began the report period at 100 percent power. On July 24, 2000, the licensee reduced power to approximately 31 percent to repair a main condenser tube leak. Following main condenser tube repairs on July 25, 2000, Unit 1 power was returned to 100 percent. Unit 2 began the report period at 100 percent power. On July 13, 2000, operators reduced power 2 percent due to main transformer overheating. After installing temporary air conditioning units on July 14, 2000, Unit 2 was restored to 100 percent power for the remainder of the report period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1RO4 Equipment Alignment (71111.04)

.1 Partial System Walkdown

a. Inspection Scope

The inspectors conducted partial inspections of the following risk significant systems to verify that they were in their proper standby alignment. The inspectors evaluated the effectiveness of the licensee's problem identification and resolution program for resolving issues which could increase event initiation frequency or impact mitigation system availability.

- Unit 2, Train A, motor-driven auxiliary feedwater system
- Unit 2 turbine-driven auxiliary feedwater system
- Unit 1 instrument air system

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Fire Area Walkdowns

a. Inspection Scope

The inspectors toured the following areas to assess the control of transient combustible materials, the material condition and lineup of fire detection and suppression systems, and the material condition of manual firefighting equipment and passive fire barriers:

- Unit 2, Train B, switchgear room (Fire Area 2SE018)
- Unit 2, 810 foot level corridor (Fire Area 2SB008)
- Auxiliary Building 790 foot elevation (Fire Area AA021)
- Unit 1, Train A, inverter and distribution room (Fire Area EH053)

The inspectors evaluated the effectiveness of compensatory measures for degraded equipment.

b. Findings

No findings were identified.

1RO6 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors conducted an annual inspection of flood protection measures. This included a review of flood analysis documentation to determine areas in the plant susceptible to flooding from internal and external sources. Based on that review and a review of the probabilistic risk analysis, a walkdown of the Units 1 and 2 safety chiller rooms was performed to assess the adequacy of flood protection measures.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

.1 (Closed) Unresolved Item 50-445(446)/200004-01: failure to identify emergency diesel generator (EDG) jacket water cooler fouling, a degraded condition beyond the recommended acceptance criteria contained in Station Administrative Manual Procedure STA-734, “Service Water System Fouling Monitoring Program,” Section 6.2.2.2 A.

The inspector followed up on an unresolved item identified during an annual inspection of the licensee’s EDG jacket water coolers fouling monitoring program as described in integrated NRC Inspection Report 50-445(446)/00-04. The inspector's followup effort included: (1) a review of the licensee’s formal operability review; (2) a determination of the significance of a violation of 10 CFR Part 50, Appendix B, Criterion XVI, for failing to identify and correct multiple degraded conditions in a timely manner; (3) a review of the test acceptance criteria; (4) a review of incorporation of instrument measurement accuracy in heat exchanger fouling calculations; and (5) a review of the effect of not incorporating margin for service water flow in fouling calculations.

Emergency Diesel Generator Operability

The licensee concluded that the EDGs remained operable throughout the entire period when the previously unidentified degraded conditions existed. For the qualitative analysis, the licensee selected the worst case fouling margin which was determined to be on the Unit 1, Train B, EDG jacket water cooler during June 1999. Since the EDG turbocharger and the engine lube oil temperatures are the limiting parameters, the operability evaluation appropriately focused on a verification that the EDG turbocharger and engine lube oil temperature would have remained within design limits during a postulated design basis event considering both the excessive fouling and the postaccident peak service water inlet temperature for the given service water temperature.

The June 1999 test data revealed that the Unit 1, Train B, EDG was loaded to 6500 kilowatts (kW) during the test, 200 kW more than required for the design basis event. The initial service water temperature was 88.8°F, 13.2°F less than that assumed in the accident analysis. The maximum design turbocharger temperature specified by the vendor was 1,200°F, and measured turbocharger temperature during the June 1999 surveillance test was 1,140°F, 60°F less than the maximum design temperature. The service water inlet temperature rise previously calculated for the postulated accident was conservatively determined to be 14°F. Assuming the temperature rise on the turbo charger proportionally changed the same amount as the service water temperature rise following a postulated accident, the maximum turbocharger temperature would be less than its design limit.

Even though the licensee's postoperability review was not a quantitative and rigorous heat balance calculation, the inspector found the qualitative argument credible.

<u>Significance of 10 CFR Part 50, Appendix B, Criterion XVI violation</u>

As discussed in NRC Inspection Report 50-445(446)/00-04, the inspector noted that trending had not been conducted as specified in STA-734 for all four EDGs during the fall of 1998, for the Unit 1 EDGs during the spring of 1999, and for the Unit 2 EDGs during the fall of 1999. As a result, the Unit 1 EDG jacket water coolers had not been trended for approximately 1 ½ years and the Unit 2 EDG jacket water coolers had not been trended for about one year. During that period, the Units 1 and 2, Train B, EDG jacket water coolers were fouled beyond the acceptance criteria contained within STA-734, Section 6.2.2.2 A, and were thus degraded, approximately 90 percent of the time. From November 1998 to June 2000, the Unit 2, Train B, EDG jacket water cooler fouling was excessive and only satisfied the acceptance criteria contained in STA-734, Section 6.2.2.2 A, for the short period between March and May 1999. These degraded conditions were not identified and placed into the licensee's corrective action program.

STA-734, Section 6.2.2.2 A states, in part, that "Test data should be analyzed in accordance with Calculation ME-CA-011-3075. . . to correlate to a fouling factor for the heat exchanger. . . This fouling factor will be compared to an acceptable fouling factor which has been calculated for 102°F SW [Service Water] temperature. If the fouling factor is determined to be unacceptable, then the affected DG [emergency diesel generator] system train should be declared INOPERABLE and the appropriate Technical Specification requirements satisfied."

10 CFR Part 50, Appendix B, Criterion XVI, states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected."

Contrary to the above, the licensee failed to promptly identify that on numerous occasions during 1998 and 1999, EDG jacket water cooler fouling had increased beyond the acceptance criteria specified in STA-734, Section 6.2.2.2 A, a condition adverse to quality.

This violation of 10 CFR Part 50, Appendix B, Criterion XVI, is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-445;446/200005-01). The issue was placed into the licensee's problem identification and resolution program as Smart Form SMF-2000-0001548-00.

A review of the past test data showed that the failure to place the degraded EDG jacket water cooler issue in the corrective action program for 2 years did not result in an EDG becoming inoperable. As such, the inspectors evaluated the resultant impact of the issue on the plant using the significance determination process and concluded that the finding resulted in a very low risk significant (green) condition adverse to quality. Additionally, the inspector concluded that the failure to recognize a degraded condition and place it into the corrective action program was an additional example of engineering quality issues identified in previous NRC inspection reports.

<u>Acceptance Criteria</u>

The inspector reviewed the acceptance criteria specified in STA-734, Section 6.2.2.2 A, to determine whether it was appropriately based on the current licensing and design bases information.

The Comanche Peak Steam Electric Station Final Safety Analysis Report (FSAR), Section 9.5.5.1, “Design Bases,” states “The emergency diesel generator jacket water cooling system is designed to allow the diesel generator sets to be rapidly loaded and to operate continuously at their maximum ratings. The various components are sized to remove the maximum heat produced by the diesel generator sets using 115°F service water as a cooling medium.” Final Safety Analysis Report Table 9.5-18, “Jacket Water Cooler Design Parameters,” lists a design heat removal rate of 25.5 X 10^6 BTU/hour, a required heat removal rate of 17.0 X 10^6 BTU/hour, and a maximum service water inlet temperature of 115°F.

The Comanche Peak Steam Electric Station Supplemental Safety Evaluation Report, Section 9.5.5, states “The staff concluded. . . that the diesel generator cooling water components are adequately sized to remove the maximum heat produced by the diesel generator sets, using 115°F service water as a cooling medium. On this basis, the staff finds that 115°F service water to the diesel generator cooling water components is acceptable.”

Comanche Peak Steam Electric Station Design Basis Document DBD-ME-011, “Diesel Generator Sets,” Revision 11, dated March 11, 1999, Section 4.3.1.1.3 states “The DGJWS [Diesel Generator Jacket Water System] shall be designed to allow the diesel generator sets to be rapidly loaded and to operate continuously at their maximum ratings. The various components are required to be sized to remove the maximum heat produced by the diesel generator sets using 115°F service water as a cooling medium.”

The inspector found that the 102°F acceptance criteria contained in STA-734, Section 6.2.2.2 A, appropriately accounted for the 13°F service water temperature rise associated with a design basis accident which is documented in Calculations 2-ME-0042, Revision 0, and ME-(B)-391.

Instrument Accuracy in Fouling Calculation

The inspector found that instrument accuracy was sufficiently considered. One instrument installed in the plant had an accuracy of ±2 percent while the engineer was assuming ±1 percent in the fouling calculations. The impact of this error was minor, considering the range for which the instrument was used, and did not affect the results of the calculation.

Service Water Flow Margin in Fouling Calculation

The inspector found that actual service water flow was used in the fouling margin calculation and was concerned about the effect of safe shutdown impoundment level changes during postulated accident conditions. The licensee's sensitivity analysis demonstrated that the service water flow would only be marginally affected during a postulated design basis accident and, as a result, no service water flow margin consideration was needed.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors observed licensed staff members during a training scenario in the control room simulator and attended the posttraining critique. Simulator observations concentrated on the conduct of operations, procedure usage, and command and control.

b. Findings

No findings were identified.

1R12 Maintenance Rule (71111.12)

.1 Maintenance Rule Functional Failure Review

a. Inspection Scope

The inspectors independently verified that the licensee properly implemented 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for the following equipment performance problems.

- Unit 1 charging pump suction solenoid-operated vent valve failure to close on demand and a subsequent change to leave the solenoids closed and de-energized with compensatory actions to periodically vent the charging pump suction piping of accumulated noncondensible gasses.

- Unit 2 containment nitrogen supply leakage and subsequent operation of the facility with the nitrogen containment isolation valve closed. Compensatory actions to reopen the containment isolation valve if nitrogen accumulators inside containment needed recharging were also reviewed.

Generally, the inspector's review focused on whether the structures, systems, or components (SSCs) that experienced problems were properly characterized with respect to the scope of the program. The review also looked at whether the SSC failure or performance problem was properly characterized and the adequacy of the licensee's significance classification for the SSC. The appropriateness of the performance criteria established for the SSC (if applicable) and the adequacy of corrective actions for SSCs classified in accordance with 10 CFR50.65 a(1) were also reviewed.

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's risk assessment for the following emergent at-power work and observed portions of the work:

- Unit 1 main condenser tube leak repairs
- Unit 1, Train B component cooling water heat exchanger cleaning
- Unit 1, Loop 4 Nitrogen 16 instrument power supply replacement

When the need for emergent work was identified on risk significant structures, systems, or components, the inspectors evaluated the licensee's actions to plan and control the resulting activities, including the acceptability of any necessary compensatory actions and contingency plans, when applicable.

b. Findings

No findings were identified.

1R14 Nonroutine Plant Evolutions and Events (71111.14)

a. Inspection Scope

On July 24, 2000, the licensee conservatively reduced power in Unit 1 to below 35 percent in response to a main condenser tube leak. Once the tube leak was repaired, the unit was restored to power. The inspectors observed operators reduce power, stabilize the plant, and then restore full power the following day.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed selected operability evaluations involving risk significant systems or components conducted by the licensee during the report period. The inspectors evaluated the technical adequacy of the licensee’s operability determination, verified that appropriate compensatory measures were implemented, and verified that the licensee considered all applicable pre-existing conditions. Additionally, the inspectors evaluated the adequacy of the licensee’s problem identification and resolution program as it applied to operability evaluations. The following specific operability evaluations were reviewed:

- Unit 2, Steam Generator 3, main steam isolation valve packing leak
- Common safety-related instrument air accumulator low pressure alarm setpoint deficiencies

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspector reviewed the results of postmaintenance testing for the following maintenance activities:

- Unit 2, Train B motor-driven auxiliary feedwater pump lantern ring inspection
- Unit 1, Loop 4 Nitrogen 16 instrument power supply replacement

For each case, the associated work orders and test procedures were reviewed to determine the scope of the maintenance activity and to determine the adequacy of the test to assure that the components affected by the maintenance were operable. The Updated FSAR, Design Basis Documents, and selected calculations were also reviewed to determine the adequacy of the acceptance criteria listed in the test procedures.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors evaluated the adequacy of selected periodic testing for important nuclear plant equipment. Aspects such as preconditioning, the impact of testing during plant operations, the adequacy of acceptance criteria, test equipment accuracy, procedure adherence, record keeping, the restoration of standby equipment, test failure evaluations, jumper control (if applicable), and the effectiveness of the licensee's problem identification and correction program. The following surveillance test activities were observed by the inspectors:

- Unit 2 turbine-driven auxiliary feedwater pump surveillance test
- Unit 2, Train A, switchgear room fire door testing
- Unit 2, Train B, motor-driven auxiliary feedwater pump surveillance test

b. Findings

No findings were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors walked down and reviewed the following temporary plant modifications:

- Unit 2, Main Transformer 1 and 2 temporary air conditioner installation
- Unit 2, Instrument Air Compressor 2-01 temporary air conditioner installation

b. Findings

No findings were identified.

1EP2 Alert and Notification System Testing (71114.02)

a. Inspection Scope

The inspector reviewed the following aspects of the licensee's alert and notification system to determine whether the siren maintenance and testing program was properly implemented:

- Vendor documentation of the original design of the alert notification system

- Vendor documentation relating to a siren system upgrade

- Emergency plan and licensing commitments pertaining to siren maintenance and testing

- Licensee procedures for performing siren maintenance and testing

b. Findings

No findings were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspector reviewed the following aspects of the licensee's system for notification and augmentation of the onsite emergency response organization to determine whether the licensee was capable of meeting their established emergency response facility staffing goals.

- Vendor documentation of the design of the emergency response organization augmentation system
- Emergency plan and licensing commitments related to the augmentation system
- Results of augmentation system drills conducted in the last 18 months
- Emergency response organization duty roster
- Primary and backup licensee procedures for initiating emergency response organization augmentation

b. Findings

No findings were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspector reviewed the following items to evaluate the licensee's ability to identify and resolve emergency preparedness related problems:

- Licensee corrective action program procedures
- Summaries of 106 Smart Forms assigned to emergency preparedness since January 1999, from drills and exercises, emergency response organization training, and a sample of 12 Smart Forms related to drills and exercises. Each of these consisted of an aggregation of many individual critique items, with some forms tracking up to 40 separate corrective actions
- A sample of 11 completed corrective actions

- The two most recent emergency preparedness program 10 CFR 50.54(t) audits and two Smart Forms related to licensee audits

- Other emergency preparedness internal and external assessments

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspector observed operator performance during a control room minidrill conducted in the simulator. Observations were focused on emergency classifications, offsite notifications, and protective action recommendation development activities. The inspector also attended the posttraining critique to compare observations against those of the licensee to determine if they were adequately identifying performance problems.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Drill and Exercise Performance

a. Inspection Scope

The inspector reviewed exercise and drill records from the second calendar quarter of 1999 and the first quarter of 2000 to verify accuracy of reported performance indicator data.

b. Findings

The inspector identified one notification opportunity that the licensee did not include in the reported performance indicator data. This opportunity resulted from an additional, unanticipated classification made in a first quarter 2000 exercise. The licensee reported the additional classification opportunity but not its associated notification opportunity. The licensee determined the additional notification opportunity was satisfactorily completed since there was no unsatisfactory performance documented in the exercise report. The inspector determined that the additional opportunity would not adversely change the performance indicator data, which remained in the licensee control band. The inspector was unable to verify the accuracy of the results reported for this opportunity since the licensee did not maintain copies of the forms that verified notification accuracy and timeliness. The licensee instead maintained performance summaries that documented numbers of opportunities and successes. The licensee entered the issue into its corrective action system as Smart Form SMF-2000-001834-00.

.2 Emergency Response Organization Drill Participation

a. Inspection Scope

The inspector reviewed drill and exercise attendance records for a sample of 25 key emergency response organization members whose drill participation was reported in the performance indicator for the first quarter of calendar year 2000. The inspector reviewed these documents to verify the accuracy of reported performance indicator (PI) data.

b. Findings

No findings were identified.

.3 Alert and Notification System Reliability

a. Inspection Scope

The inspector reviewed siren testing printouts and summary results for the first calendar quarter of 2000 to verify the accuracy of reported PI data for this quarter.

b. Findings

No findings were identified.

.4 Mitigating Systems

a. Inspection Scope

The inspectors conducted a review of the licensee's Units 1 and 2 mitigating systems PI data for the first quarter of 2000 to determine its accuracy and completeness.

b. Findings

No findings were identified.

40A6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. Mike Blevins and other members of licensee management at an exit meeting on July 18, 2000. The licensee acknowledged the findings presented. The licensee stated during the exit meeting that they believed the potential violation discussed in Section 1R07 associated with a failure to identify the degraded condition involving excessive fouling of the EDG jacket water coolers would more appropriately be cited against 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings." The licensee's position was that, since the degraded EDG jacket water cooler fouling monitoring program described in Procedure STA-734 was not properly implemented, a potential violation of Criterion V would more appropriately reflect the procedure adherence issue. The inspector explained that, since Procedure STA-734 was developed in response to several commitments made while responding to NRC Generic Letter 89-13, the procedural steps were not regulatory requirements nor were they worded as such and citing Criterion V would be incorrect. The inspector also explained that citing Criterion XVI was more appropriate because it focused on the consequence of not properly implementing the monitoring program.

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

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Blevins, Vice President, Nuclear Operations
M. Bozeman, Emergency Preparedness Manager
N. Hood, Executive Assistant
T. Hope, Regulatory Compliance Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-445(446)/200006-01	NCV	Failure to promptly identify and correct degraded EDG jacket water coolers.

Closed

50-445(446)/200006-01	NCV	Failure to promptly identify and correct degraded EDG jacket water coolers.
50-445(446)/200004-01	URI	Failure to identify EDG jacket water cooler fouling.

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Station Administration Manual Procedures:

- STA-734, “Service Water System Fouling Monitoring Program,” Revisions 0, 1, and 2

Miscellaneous Procedures:

- TSP-503, “Emergency Diesel Generator Reliability Program," Revision 3

Calculations:

- ME-CA-0011-3075, Revision 1, “Emergency Diesel Generator Jacket Water Cooler Fouling Factor Analysis”

- ME-(B)-391, Revision 2, “Service Water Flow to Emergency Diesel Generator Jacket Water Cooler”

- 2-ME-0042, Revision 0, “Minimum Allowable Service Water Flow to the Emergency Diesel Generator”

Smart Forms:

- SMF-2000-001548-00, Failure to Properly Implement Diesel Generator Jacket Water Cooler Fouling Monitoring Program
- SMF 1999-001426-00 to 15
- SMF 1999-001431-00 to 37
- SMF 1999-001893-00 to 18
- SMF 1999-002043-00 to 43
- SMF 1999-002095-00
- SMF 1999-002098-00 to 02
- SMF 2000-000106
- SMF 2000-000514-00 to 24
- SMF 2000-001051-00 to 40
- SMF 2000-001439-00 to 42
- SMF 2000-001528
- SMF 2000-001529
- SMF 2000-001531
- SMF 2000-001799
- SMF 2000-001834-00

Design Basis Documents:

- DBD-ME-011, "Diesel Generator Sets," Revision 11

Quick Technical Evaluations:

- QTE-2000-1864-01-00, safety-related instrument air accumulator operability evaluation
- QTE-2000-001812-01-00, operability evaluation for main steam isolation valve packing steam leak

<u>Emergency Preparedness Procedures</u>:

EPP-100	“Maintaining Emergency Preparedness”	Revision 3
EPP-203	“Notifications”	Revision 13
EPP-204	“Activation and Operation of the Technical Support Center”	Revision 14
EPP-205	“Activation and Operation of the Operations Support Center”	Revision 11
EPP-206	“Activation and Operation of the Emergency Operations Facility”	Revision 14
ODA-102	“Conduct of Operations”	Revision 19
MSE-PO-9329	“Emergency Alerting System Inspection”	Revision 3
SEC-610	“Security Response during Personnel and Operating Emergencies”	Revision 13
SG-012	“Offsite Alerting System (Sirens) Surveillance”	Revision 3
STA 421	“Initiation and Processing of Smart Forms”	Revision 8
STA 422	“Disposition of Smart Forms Identifying Potential Adverse” Conditions”	Revision 15
SG-020	“NRC Performance Indicators”	Revision 0
TRA-105	“Emergency Preparedness Training”	Revision 17
Staff Guidance 05	“Quarterly Contact Verification of the Emergency Response Organization”	Revision 7
Staff Guidance 12	“Offsite Alerting System (Sirens) Surveillances”	Revision 3

<u>Self-Assessments</u>:

- Log 99-001, End of the Year Documentation Review, dated 1/10/99
- Log 99-004, Emergency Plan versus Emergency Plan Commitments Comparison HPES 99-1301, Nuclear Overview HPES Report
- Log 99-006, Review of 1999 Emergency Response Organization Training, dated January 7, 2000
- EVAL-2000-003, Emergency Plan Revision 28 Change Implementation Evaluation, January 10-19, 2000

- Log 99-008, Engineering Workbook Review, dated January 19, 2000

- Log 99-007, Rad Workbook Review, dated January 25, 2000

- NOD Evaluation EVAL-1999-059, “Regulatory Affairs Self-Assessment and NOD Evaluation to Validate and Verify Regulatory Assessment Performance Indicator Data,” dated January 24, 2000

- EVAL-2000-024, Emergency Planning Evaluation/Self Assessment, May 17 through June 1, 2000

- Indian Point 2 SGTR and Alert Industry Event Review, dated July 10, 2000

- Log 99-005, Signs and Posters Inspection, dated July 17, 2000

- Log 2000-01, Emergency Plan versus Its Commitments

<u>Drill Reports</u>:

- May 18, 1999, Red Team Exercise
- June 8, 1999, Green Team Dress Rehearsal
- July 21, 1999, Green Team Biennial Exercise
- August 20, 1999, Blue Team Exercise
- March 1, 2000, Red Team Exercise
- May 20, 2000, Green Team Exercise

<u>Correspondence related to Siren Systems</u>:

- Gary E. Jones, FEMA Region VI, to Al Armstrong, TUG Company, dated January 8, 1985

- Al Lookabaugh, FEMA Region VI, to Al Armstrong, TUG Company, dated February 12, 1985

- Jerry D. Stephens, FEMA Region VI, to Robert A. Lansford, Texas Department of Public Safety, Division of Emergency Management, dated August 9, 1985

- Robert A. Lansford, Texas Department of Public Safety, Division of Emergency Management, to Jay Laughlin, TUG Company, dated August 19, 1985

- Jerry D. Stephens, FEMA Region VI, to Robert A. Lansford, Texas Department of Public Safety, Division of Emergency Management, dated April 28, 1986

- Robert A. Lansford, Texas Department of Public Safety, Division of Emergency Management, to Dell Greer, FEMA Region VI, dated December 10, 1986

- Tom L. Gosdin, TUG Company, to Robert A. Lansford, Texas Department of Public

Safety, Division of Emergency Management, dated December 3, 1986

- Robert A. Lansford, Texas Department of Public Safety, Division of Emergency Management, to Al Armstrong, TUG Company, dated March 15, 1988

- A. B. Scott, TU Electric to Robert A. Lansford, Texas Department of Public Safety, Division of Emergency Management, dated July 1, 1988

Nuclear Commitment Compliance Report:

- Commitment #02043
- Commitment #06297
- Commitment #06771
- Commitment #14599
- Commitment #14604
- Commitment #14611
- Commitment #15617
- Commitment #27100

Other Documents:

- Comanche Peak Steam Electric Station, Units 1 and 2, Technical Specifications

- Comanche Peak Steam Electric Station Final Safety Analysis Report

- Comanche Peak Steam Electric Station Fire Protection Report

- NRC Inspection Report 50-445(446)/90-38, Inspection of Licensee Implementation of NRC
- Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment”

- Final Report of the Alert and Notification System for Comanche Peak Steam Electric Station, dated June 8, 1984

- Software User Manual, ATI REACT-2000, Revision 2, dated March 1994

- Installation and User Manual, ATI REACT-2000 Alert and Notification System, Revision 2, dated March 1994

- The Communicator, Automated Emergency Notification System, System Administrator Guide

- Service Manual, Public Notification System, Federal Signal Corporation (Project D7275), dated June 29, 1982

- Position Assistance Document: Control Room Communicator

- Position Assistance Document: Security Communicator
- Quarterly Callout Machine and Pager Test Summary, dated January 24, 2000
- Memorandum: Management Expectations for ERO Personnel, dated August 6, 1999

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
·Initiating Events	·Occupational	·Physical Protection
·Mitigating Systems	·Public	
·Barrier Integrity		
·Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC used two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the significance determination process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, or RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.