UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

November 6, 2006

Mike Blevins, Senior Vice President
and Chief Nuclear Officer
TXU Power
ATTN: Regulatory Affairs
Comanche Peak Steam Electric Station
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000445/2006004 AND 05000446/2006004

Dear Mr. Blevins:

On September 23, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Comanche Peak Steam Electric Station, Units 1 and 2 facility. The enclosed integrated inspection report documents the inspection findings which were discussed on October 3, 2006, with Mr. R. Flores and other members of your staff.

This inspection examined activities conducted under your licenses as they related to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents one self-revealing finding of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a noncited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the NCV in this report, 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 Region IV; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at Comanche Peak Steam Electric Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made 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/reading-rm/adams.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/

Claude Johnson, Chief
Project Branch A
Division of Reactor Projects

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

Enclosure: NRC Inspection Report 05000445/2006004 and 05000446/2006004
w/Attachment: Supplemental Information

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SUNSI Review Completed: _CEJ__ ADAMS: √ Yes ☐ No Initials: __CEJ____
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-445, 50-446
Licenses:	NPF-87, NPF-89
Report:	05000445/2006004 and 05000446/2006004
Licensee:	TXU Generation Company LP
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56, Glen Rose, Texas
Dates:	June 24, 2006 through September 23, 2006
Inspectors:	D. Allen, Senior Resident Inspector A. Sanchez, Resident Inspector R. Lantz, Senior Emergency Preparedness Inspector S. Rutenkroger, Regional Inspector W. Johnson, Contractor
Approved by:	Claude Johnson, Chief, Project Branch A Division of Reactor Projects
Attachment:	Supplemental Information

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SUMMARY OF FINDINGS

IR 05000445/2006004, 05000446/2006004; 06/24/2006-09/23/2006; Comanche Peak Steam Electric Station, Units 1 and 2. Problem Identification and Resolution.

This report covered a 3-month period of inspection by two resident inspectors, one emergency preparedness inspector, one engineering inspector, and one consultant. One Green finding, which was determined to be a noncited violation, was identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using the Inspection Manual Chapter 0609, " Significance Determination Process." Findings for which the Significance Determination Process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A self-revealing noncited violation of Technical Specification 5.4.1.a was identified for I&C technicians disabling both channels of P4 Reactor Trip Interlock in Unit 1, without procedural guidance, while performing main turbine stop/control valve leakage testing in Mode 3. This resulted in the turbine unexpectedly speeding up from 74 rpm to 1800 rpm within one minute. The operators attempted to trip the turbine via the turbine trip pushbutton, but the trip push-button, as well as the P4 Reactor Trip Interlock was disabled. The operators eventually closed the control valves by setting the startup/load limit device to zero percent. The licensee entered the issue into their corrective action program.

  This finding is more than minor because the procedural error caused a transient in Mode 3 that resulted in the main turbine speeding up to 1800 rpm and a RCS cooldown from 511 degrees F to 499 degrees F. In addition, the finding affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of a system that responds to initiating events to prevent undesirable consequences. This finding is of very low safety significance in accordance with Phase 1 of Manual Chapter 0609, Appendix A because it was not a design or qualification deficiency, did not represent a loss of system safety function nor an actual loss of safety function, and did not screen as potentially risk significant due to external events. The cause of this finding is related to the crosscutting area of Human Performance because the licensee did not effectively communicate expectations regarding procedural compliance and personnel to follow procedures. (Section 4OA2.4)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Comanche Peak Steam Electric Station (CPSES) Units 1 and 2 operated at essentially 100 percent power for the entire reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors: (1) walked down portions of the below listed risk important systems and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walkdown to the licensee's corrective action program to ensure problems were being identified and corrected.

- Unit 2 Motor Driven Auxiliary Feedwater Pump 2-02 in accordance with Operations Testing Manual (OPT) Procedure OPT-206B, "AFW System," Revision 18, and System Operating Procedure (SOP) Manual SOP-304B, "Auxiliary Feedwater System," Revision 10, while Motor Driven Auxiliary Feedwater Pump 2-01 was inoperable for scheduled surveillance testing on July 27, 2006

- Unit 2 Containment Spray Pumps 2-01 and 2-03 in accordance with OPT-205B, "Containment Spray System," Revision 13, and SOP-204B, "Containment Spray System," Revision 5, while Containment Spray Pumps 2-02 and 2-04 were inoperable due to planned motor operated valve inspections, on August 1, 2006

- Unit 1 residual heat removal system Train B in accordance with procedures OPT-203A, "Residual Heat Removal System," Revision 15, and SOP-102A, "Residual Heat Removal System," Revision 15, while Train A residual heat removal system was inoperable for scheduled maintenance and surveillance testing on August 17, 2006

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Fire Area Tours (71111.05Q)

a. Inspection Scope

The inspectors walked down the listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features; and (7) reviewed the corrective action program to determine if the licensee identified and corrected fire protection problems.

- Fire Zone AA21D - Units 1 and 2 auxiliary building 830' elevation on July 17, 2006
- Fire Zone AA21C - Units 1 and 2 auxiliary building 822' elevation, Rooms 208 and 209, chemical volume control system alternate mini-flow valve area on July 19, 2006
- Fire Zone WB104a/b - Units 1 and 2 station service water intake structure on July 21, 2006
- Fire Zone AA96, 97, 99D, and 99E - fuel handling building on July 25, 2006
- Fire Zone SD-009 - Unit 1 Train A switchgear room on August 15, 2006
- Fire Zone SG-010 - Unit 1 Train A emergency diesel generator room on August 15, 2006

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill (71111.05A)

a. Inspection Scope

The inspector observed two fire brigade drills, performed on March 8, 2006 and August 15, 2006, to evaluate the readiness of licensee personnel to prevent and fight fires, including the following aspects: (1) use of protective clothing; (2) use of breathing apparatuses; (3) placement and use of fire hoses; (4) entry into the fire area; (5) use of firefighting equipment; (6) brigade leader command and control; (7) communications between the fire brigade and control room; (8) searches for fire victims and fire propagation; (9) use of fire pre-plans; (10) adherence to the drill scenario; and (11) the drill critique. The licensee simulated a fire in the Unit 2 Train B 1E Battery Room, and among the temporary Bechtel trailers near the fire pump house, respectively.

The inspector completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

The inspector observed a licensed operator requalification training scenario in the control room simulator on July 17, 2006. This scenario used the new software and a temporary control board modification to simulate the new steam generators that Unit 1 will be installing Spring 2007. The scenario began with a short event to take initial operator action for the loss of a heater drain pump. The main scenario began with operators taking the watch with the reactor at 100 percent power. The following events then took place: (1) a low pressure heater bypass valve failed open and the crew was allowed to recover axial flux difference within specification; (2) a main feedwater pump speed control failed high; (3) the main generator automatic voltage control regulator failed; (4) main generator vacuum leak which required a manual reactor trip; and (5) a partial safety injection signal.

Simulator observations included formality and clarity of communications, group dynamics, the conduct of operations, procedure usage, command and control, and activities associated with the emergency plan. The inspectors also verified that evaluators and the operators were identifying crew performance problems as applicable.

The inspector attended a classroom session, on July 17, 2006, concerning the main steam system and how it will be modified for the steam generator replacement that will occur during the Unit 1, Spring 2007, refueling outage.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors independently verified that CPSES personnel properly implemented 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” for the following equipment performance items:

- The Buildings and Structures system, which included Units 1 and 2 buildings as well as common buildings, has Maintenance Rule Functions and was included in the scope of the Maintenance Rule program. This system was selected due to a number of long term corrosion problems which have persisted for years.

- Recent failures and maintenance issues with the fire protection system diesel engines for Fire Pumps X-05 and X-06 which resulted in increased unavailability although the fire protection system recently transitioned from Maintenance Rule (a)(1) status to (a)(2) status.

The inspectors reviewed whether the structures, systems, or components (SSCs) that experienced problems were properly characterized in the scope of the Maintenance Rule Program and whether the SSC failure or performance problem was properly characterized. The inspectors assessed the appropriateness of the performance criteria established for the SSCs where applicable. The inspectors also independently verified that the corrective actions and responses were appropriate and adequate.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed selected activities regarding risk evaluations and overall plant configuration control. The inspectors discussed emergent work issues with work control personnel and reviewed the potential risk impact of these activities to verify that the work was adequately planned, controlled, and executed. The activities reviewed were associated with:

- The rescheduling of several work activities during the week of July 17-21, 2006 due to extremely warm weather that caused the Transmission Grid Manager (TGM) to request a “hands off” for the period

- Performing a postmaintenance test that caused the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) to be inoperable during a “hands off” request from the Qualified Scheduling Entity (QSE) for economic reasons on August 8, 2006

- Postponement of Unit 2 TDAFWP run due to a “hands off” request from the Electric Reliability Council of Texas (ERCOT) on August 14-18, 2006

- The unplanned ERCOT issuance of an “alert” and “hands off” notices due to extremely warm temperatures and resulting low reserve margin conditions on the Texas grid on September 1, 2006

- Scheduling emergent work on Safety Chiller Recirculation Pump Motor Breaker 1-05 on September 12, 15, and 18, 2006. An emergent limit switch adjustment on the Turbine Driven Auxiliary Feedwater Pump was rescheduled to avoid the Safety Chiller emergent work on September 15, 2006

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the Updated Safety Analysis Report and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any Technical Specifications; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components. The inspectors interviewed appropriate licensee personnel to provide clarity to operability evaluations, as necessary. Specific operability evaluations reviewed are listed below:

- Evaluation (EVAL) EVAL-2006-002903-01-00, determine past and present operability of Chilled Water Recirculation Pump 1-05 motor breaker 1EB2-1/10M/BKR after it experienced elevated temperatures in the termination block on September 11-15, 2006, captured in Smart Form (SMF) SMF-2006-002903, and reviewed on September 18, 2006

- EVAL-2003-000188-04-00 was updated on September 15, 2006 to determine operability of the current existing atmospheric relief valve block valve due to seismic concerns, and reviewed on September 20, 2006

- SMF-2006-002321-00, non-Q circuit board found to be used in Inverter CP1-ECIVEC-02, a safeguards balance of plant inverter, during preventive maintenance, reviewed on September 20, 2006

- SMF-2006-002950-00, loose fastener found on 1-FE-4472-1, the flange for the Unit 1 Containment Spray Pump 1-01 flow element, reviewed on September 20-21, 2006

- SMF-2006-002952-00, missing fastener on CP1-CTEDCS-03, the chemical addition eductor associated with Unit 1 Containment Spray Pump 1-03, reviewed on September 20-21, 2006

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors witnessed or reviewed the results of the postmaintenance tests for the following maintenance activities:

- Unit 2 Train B Containment Spray Valves 2-HV-4777, 2-FV-4773-1, and 2-FV-4773-2 following major motor operator inspections, in accordance with OPT-205B, “Containment Spray System,” Revision 13, on August 1, 2006

- Spent Fuel Pool Cooling Pump X-02 following rework/replacement of the rotating element and bearings, in accordance with Equipment Test Procedure ETP-209, “Spent Fuel Pool Pump Test,” Revision 0 to create a new pump curve and reference value in accordance with the ASME OM Code, and OPT-223, “Spent Fuel Pool Cooling System,” Revision 9, on August 5, 2006

- Unit 1 TDAFWP Steam Generator 1-01 flow control valve controller 1-FK-2459B tracking driver card replacement, in the remote shutdown panel, in accordance with OPT-216A, “Remote Shutdown Operability Test,” Revision 10, on August 9, 2006

- Unit 1 Train A residual heat removal system following inspection of motor operated actuator for Mini-Flow Valve 1-FCV-610, diagnostic testing of suction Valve 1-8812A, pump breaker maintenance, and lubrication oil change for pump and motor, in accordance with OPT-203A, “Residual Heat Removal System,” Revision 15, on August 17, 2006

- Unit 1 turbine driven auxiliary feedwater pump and steam supply Valve 1-HV-2452-2 from Main Steam Line 1 following the replacement of a solenoid valve, in accordance with OPT-603A, “TDAFW Accumulator Check Valve Leak Test,” Revision 5, and OPT-206A, “AFW System,” Revision 25, on August 25, 2006

- Unit 1 Safety Chilled Water Recirculation Pump Motor Breaker 1EB3-1/10M/COMP following the replacement of the terminal block and the splicing of new cable replacing approximately one foot of “B” and “C” phase conductors on the end entering the terminal block, in accordance with work order WO-4-06-169766-00 on September 18, 2006

In each case, the associated work orders and test procedures were reviewed in accordance with the inspection procedure to determine the scope of the maintenance activity and to determine if the testing was adequate to verify equipment operability.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors evaluated the adequacy of periodic testing of important nuclear plant equipment, including aspects such as preconditioning, the impact of testing during plant operations, and the adequacy of acceptance criteria. Other aspects evaluated included test frequency and test equipment accuracy, range, and calibration; procedure adherence; record keeping; the restoration of standby equipment; test failure evaluations; system alarm and annunciator functionality; and the effectiveness of the licensee’s problem identification and correction program. The following surveillance test activities were observed and/or reviewed by the inspectors:

- Unit 1 Containment Spray Pumps 1-01 and 1-03 in accordance with OPT-205A, “Containment Spray System,” Revision 15, observed on June 27, 2006

- Unit 2 turbine driven auxiliary feedwater pump in accordance with OPT-206B, “AFW System,” Revision 18, observed on July 6, 2006

- Unit 2 Containment Spray Pumps 2-01 and 2-03 in accordance with OPT-205B, “Containment Spray System,” Revision 13, observed on August 1, 2006

- Unit 1 auxiliary feedwater flow control valves from the remote shutdown panel in accordance with OPT-216A, “Remote Shutdown Operability Test,” Revision 10, reviewed on August 10, 2006

- Unit 2 Train A Centrifugal Charging Pump 2-01 in accordance with OPT-201B, “Charging System,” Revision 7, observed on August 13, 2006

- Units 1 and 2 Diesel Driven Fire Pump X-06 operability test in accordance with OPT-220, "Fire Suppression Water System Operability Test," Revision 9, observed on August 29, 2006

- Units 1 and 2 Diesel Driven Fire Pump X-05 operability test in accordance with OPT-220, "Fire Suppression Water System Operability Test," Revision 9, observed on August 29, 2006

- Unit 1 containment air particulate/iodine/gas radiation Channel 1-RE-5502/03/66 in accordance with Instrument and Control Manual (INC) procedure INC-7096, "Channel Operational Test and Channel Calibration Containment Particulate, Iodine, and Gas Channels 1-RE-5502/5503/5566 and 2-RE-5502/5503/5566," Revision 5, reviewed September 1, 2006

- Unit 2 containment air particulate/iodine/gas radiation Channel 2-RE-5502/03/66 in accordance with procedure INC-7096, "Channel Operational Test and Channel Calibration Containment Particulate, Iodine, and Gas Channels 1-RE-5502/5503/5566 and 2-RE-5502/5503/5566," Revision 5, reviewed September 1, 2006

The inspectors completed nine samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, plant drawings, procedure requirements, Technical Specification and Technical Requirements Manual to ensure that the below listed temporary modification was properly implemented. The inspectors: (1) verified that the modification did not have an affect on system operability/availability; (2) verified that the installation was consistent with the modification documents; (3) ensured that the post-installation test results were satisfactory and that the impact of the temporary modification on permanently installed SSCs were supported by the test; (4) verified that the modification was identified on control room drawings and that appropriate identification tags were placed on the affected equipment; and (5) verified that appropriate safety evaluations were completed. The inspectors verified that licensee identified and implemented any needed corrective actions associated with temporary modification.

- Unit 1 Control Rod Drive Motor Vent Fan 1-01 disconnect switch CP1-VAFNCB-01D bypass using 4/0 AWG cable as a jumper from Junction Box JB1C-412O to connector 14, reviewed on September 19, 2006

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert Notification System Testing (71114.02)

a. Inspection Scope

The inspector discussed with licensee staff the status of the offsite siren system to determine the adequacy of licensee methods for testing the alert and notification system in accordance with 10 CFR Part 50, Appendix E. The licensee's alert and notification system testing program was compared with criteria in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1; Federal Emergency Management Agency (FEMA) Report REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants"; and the licensee's current FEMA-approved alert and notification system design report. The inspector also reviewed the references listed in the attachment to this report.

The inspector completed one sample during this inspection.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspector interviewed members of the emergency planning staff responsible for training and testing of the emergency response organization. The inspector discussed with licensee staff the status of primary and backup systems for augmenting the on-shift emergency response staff to determine the adequacy of licensee methods for staffing emergency response facilities. The inspector reviewed the results of seven quarterly augmentation/notification drills. The inspector evaluated drill performance and training implementation against emergency plan implementation procedures and other documents related to the emergency response organization augmentation system to determine the licensee personnel's ability to staff emergency response facilities in accordance with the Emergency Plan, Emergency Planning Staff Guideline 5, "Quarterly Augmentation Verification of the Emergency Response Organization (ERO)," Revision 11, and the requirements of 10 CFR Part 50, Appendix E. The inspector observed a table-top security-related training drill and a drill in the control room simulator which was evaluated for emergency preparedness performance indicators. The inspector also reviewed the references listed in the attachment to this report.

The inspector completed one sample during this inspection.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspector reviewed a summary of all corrective action program documents (Smart Forms) associated with emergency preparedness generated between October 2004 and July 2006 to determine the licensee's ability to identify and correct problems in accordance with the requirements of 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E. The inspector also reviewed 2 exercise reports, 6 self-assessments, 2 quality assurance audits, 25 specific Smart Forms, and other documents listed in the attachment to this report. Corrective actions were evaluated against the requirements of Station Administrative Procedure STA-421, "Initiation and Processing of Smart Forms," Revision 12. The inspector also reviewed other documents listed in the attachment to this report.

The inspector completed one sample during this inspection.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The resident inspectors observed the conduct of three shift operating crews during simulator-based training evolutions on July 27 and August 3, 2006, which had been identified by the licensee as evolutions that would contribute to the Drill/Exercise Performance (DEP) and Emergency Response Organization (ERO) performance indicators (PI). The inspectors: (1) observed the training evolutions to identify any weaknesses and deficiencies in classification, notification, and Protective Action Requirements (PAR) development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee was properly identifying failures; and (3) determined whether licensee performance was in accordance with the guidance of the Nuclear Energy Institute (NEI) NEI 99-02 document's acceptance criteria.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Emergency Preparedness

The inspector reviewed licensee evaluations for the three emergency preparedness cornerstone performance indicators of Drill and Exercise Performance, Emergency Response Organization Participation, and Alert and Notification System Reliability for the period from October 1, 2005, through June 30, 2006. The definitions and guidance of NEI 99-02, “Regulatory Assessment Indicator Guideline,” Revision 3, and Emergency Planning Staff Guideline 20, “NRC Performance Indicators,” Revision 10, were used to verify the accuracy of the licensee’s evaluations for each performance indicator reported during the assessment period.

The inspector reviewed a 100 percent sample of drill and exercise scenarios and licensed operator simulator training sessions, notification forms, and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspector reviewed the qualification, training, and drill participation records for a sample of 10 emergency responders. The inspector reviewed alert and notification system testing procedures, maintenance records, and a 20 percent sample of siren test records. The inspector also reviewed other documents listed in the attachment to this report.

The inspector completed 3 samples during the inspection.

b. Observations

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for followup, the inspectors performed a routine screening of all items entered into the licensee’s corrective action program. This review was accomplished by reviewing the licensee’s computerized corrective action program database SMFs, reviewing hard copies of selected SMFs and attending related meetings such as Plant Event Review Committee (PERC) meetings, and Smart Form Screening Review Board (SRB).

b. Findings

No findings of significance were identified.

.2 Annual Sample Review

a. Inspection Scope

The inspector selected 25 Smart Forms for detailed review based on their linkage to event classification, notification of offsite authorities, and processes for providing protective action recommendations. The Smart Forms were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspector evaluated the Smart Forms against the requirements of Station Administrative Procedure STA-421, "Initiation and Processing of Smart Forms," Revision 12.

b. Findings and Observations

No findings of significance were identified.

.3 Selected Issue Followup - Cumulative Effects of Operator Workarounds

a. Inspection Scope

The inspectors reviewed the cumulative effects of the operator workarounds to determine: (1) the reliability, availability, and potential for misoperation of a system; (2) if multiple mitigating systems could be affected; (3) the ability of operators to respond in a correct and timely manner to plant transients and accidents; and (4) if the licensee has identified and implemented appropriate corrective actions associated with operator workarounds.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.4 Selected Issue Followup -Discovered Both Unit 1 Channels of Reactor Trip P4 Interlock Disabled in Mode 3

a. Inspection Scope

The inspectors selected this issue for an in-depth review because it challenged operator performance, was assigned a level 2 significance in the licensee's corrective action program, received an apparent cause analysis, and appeared to be a violation of NRC requirements. The inspectors reviewed the reportability and operability evaluations, reviewed the apparent cause analysis for the attributes in Inspection Procedure 95001, and compared the corrective actions to the identified causes of the event.

b. Findings

Introduction: A Green self-revealing noncited violation (NCV) of Technical Specification 5.4.1.a was identified for I&C technicians disabling both channels of P4 Reactor Trip Interlock in Unit 1 without procedural guidance while performing main turbine stop/control valve leakage testing in Mode 3.

Description: On October 8, 2005, during plant shutdown and cooldown for refueling outage 1RF11, the licensee was establishing the initial conditions to perform Equipment Test Procedure ETP-410A, "Main Turbine Stop/Control Valve Leakage Test," Revision 4. A prerequisite step in this procedure was vague in directing "ensure the plant is in a condition where no outstanding items will prevent conduct of this test nor invalidate the test results." Without procedural guidance, technicians opened the turbine trip system breakers which disabled all electrical turbine trips, including those from P4 and the manual pushbuttons. This action caused Unit 1 to be in Mode 3 with neither of the required two channels of P4 operable, contrary to the requirements of Technical Specification 3.3.2 function 8 a. This action also disabled the manual turbine trip, which was a required contingency for the test in case the turbine received steam past the control valves and began to spin up.

At 2:50 p.m. the startup/load limit device was raised to 68 percent to open the turbine stop valves. The control valves unexpectedly opened approximately 10 percent, causing the turbine speed to increase from 74 rpm on the turning gear to approximately 1829 rpm within one minute. The operators unsuccessfully attempted to trip the turbine with the turbine trip push-button before closing the turbine control valves by setting the startup/load limit device to zero percent.

Prior to this test, a clearance had tagged out control power for main transformer main trip breaker Auxiliary Relay 52bx-1b/E3, which caused the turbine control system to switch to load control and set a speed demand for 1800 rpm to the speed controller. This had no immediate effect on the turbine since the turbine trip fluid was depressurized by the turbine trip signal. When the turbine valve leakage test was begun, the turbine trip was reset, re-pressurizing the turbine trip fluid circuit.

The performance deficiency was the technicians opening the turbine trip system breakers which disabled the P4 interlock and the manual pushbuttons without procedural guidance. This operation of plant equipment was contrary to Operations Department Administration Procedure ODA-410, "System Status Control," Revision 11. The control room operators did not recognize that this action disabled P4 and their ability to trip the turbine. The Apparent Cause analysis identified communications between the operator and technicians as the "apparent cause," and "extent of cause" to include lack of clear procedure prerequisites, lack of understanding by both the operators and technicians of the effects of the actions taken, and the integrated relationship between the turbine trip system and the P4 interlock. Corrective actions included: (1) modifying the turbine control system, (2) revising the test procedure, including adding a prerequisite that the reactor trip breakers are closed, (3) reviewing all operations procedures for vague prerequisites, (4) conducting pre-outage tabletop meetings between operations management and outage control, (5) training of technicians on ODA-410 requirements and importance of clear communications, and (6) current events training for operations on the modifications and changes to the procedures.

Analysis: The disabling of the turbine trips and P4 interlocks by I&C technicians without procedural guidance was the performance deficiency. Manual Chapter 0612, Appendix E, section 4, "Insignificant Procedural Errors", example b. was reviewed and provides an example of a procedural error that would not be minor if "the error caused a reactor trip or other transient." This finding was more than minor because this procedural error caused a transient in Mode 3 that resulted in the turbine speed increasing from 74 rpm to over 1800 rpm in less than a minute and the reactor coolant system cooling down from 511 degrees F to 499 degrees F in the same time. The purpose of the P4 permissive in Mode 3 is to ensure the turbine will trip and main feedwater will isolate if the reactor trip breakers open in order to limit RCS cooldown and the corresponding positive reactivity addition. In addition, this finding is also associated with the Mitigating Systems cornerstone attribute of Human Performance, Human Error (pre-event) and affects the objective to ensure the availability, reliability, and capability of a system that responds to initiating events to prevent undesirable consequences.

The significance of the finding was determined to be of very low safety significance (Green) in accordance with Phase 1 of Manual Chapter 0609, Appendix A because the finding degraded reactivity control under the Mitigating Systems cornerstone, but was not a design or qualification deficiency, did not represent a loss of system safety function nor an actual loss of safety function, and did not screen as potentially risk significant due to external events. Although the event occurred while the reactor was shutdown in Mode 3, Appendix G of Manual Chapter 0609 was not used since it only applies when the plant has met the entry conditions for cooling with the residual heat removal system. The degradation of reactivity control was limited by the RCS boron concentration meeting the shutdown margin requirements and by the reduced decay heat due to the time after shutdown of 6 hours.

This finding has a cross-cutting aspect in the area of human performance because the licensee did not effectively communicate expectations regarding procedural compliance and personnel to follow procedures.

Enforcement: Technical Specification 5.4.1.a requires that procedures shall be established, implemented, and maintained covering activities described in Regulatory Guide 1.33, Revision 2, Appendix A, which includes administrative procedures for authorities and responsibilities for safe operation. ODA-410, "System Status Control," Revision 11, specified that plant components should only be operated as directed by approved procedures or other authorized processes. Contrary to the above, on October 8, 2005, I&C technicians disabled both required channels of the reactor trip P4 interlock in Mode 3 without procedural guidance. This violation was entered into the licensee's corrective action program as Smart Forms SMF-2005-3924, SMF-2005-3925, and SMF-2005-3926. Because this finding is of very low safety significance and was entered into the licensee's corrective action program, it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000445/2006004-01, Both Unit 1 Channels of Reactor Trip P4 Interlock Disabled in Mode 3.

4OA3 Event Followup (71153)

(Closed) Licensee Event Report 50-00446/2005-003-00, Two Pressurizer Safety Valves Found With Unsatisfactory Lift Setpoints

On April 27, 2005, the licensee discovered that Unit 2 pressurizer safety Valves 2-8010A and 2-8010B failed the as found lift pressure setpoint surveillance test performed by an offsite contractor by lifting at a pressure of 1.4 and 1.3 percent, respectively, below the Technical Specification 3.4.10.1 acceptable setpoint range of 2460 to 2510 psig (2485 psig +/-1%). During review of this event, the licensee found that during the previous refueling outage, Valves 2-8010B and 2-8010C had also failed their surveillance testing by an offsite contractor, lifting 2.8 and 1.6 percent below the acceptable setpoint range in October 2003. The reason for the failures was determined to be inability of the valves to repetitively perform within the restrictive tolerance required by the Technical Specification acceptance criteria. The deviations were considered to be within the design requirements of the valves and did not indicate a material problem with the valves. The pressurizer safety valves were reworked by the vendor, the technical specification surveillance requirements were successfully met, and the valves were reinstalled in the plant. The licensee determined that the valve setpoints remained within the analyzed range in the accident analysis and that the valves remained capable of fulfilling their safety function. The licensee planned to evaluate a possible change to the Technical Specification allowed setpoint tolerance. The licensee event report was reviewed by the inspectors and no findings of significance were identified. This event constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the Enforcement Policy. The licensee documented the issue in SMF-2005-001666-00. This licensee event report is closed.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The inspector presented the emergency preparedness program baseline inspection results to Mr. M. Kanavos, Plant Manager, and other members of licensee management at the conclusion of the inspection on July 28, 2006. The licensee acknowledged the findings presented. The inspector verified no proprietary information was discussed during the inspection.

On October 3, 2006, the inspectors presented the resident inspection results to Mr. R. Flores, Vice President Nuclear Operation, and other members of licensee management. The inspectors confirmed that proprietary information was not provided during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Blevins, Senior Vice President and Chief Nuclear Officer
D. Bozeman, Manager, Emergency Planning
S. Bradley, Supervisor, Health Physics, Radiation Protection & Safety Services
T. Clouser, Manager, Shift Operations
J. Curtis, Radiation Protection Manager, Radiation and Industrial Safety
K. Faver, Emergency Planning Analyst
R. Fishencord, Emergency Planning Analyst
R. Flores, Vice President, Nuclear Operations
J. Gallman, Senior Nuclear Analyst (Work Week Coordinator)
B. Henley, Engineering Consultant (Seismic Analysis)
D. Holland, Senior Nuclear Analyst (Work Week Coordinator)
M. Kanavos, Plant Manager
S. Karpyak, Risk & Reliability Engineering Supervisor
R. Kidwell, Sr. Nuclear Technologist, Regulatory Affairs
D. Kross, Director, Maintenance
J. Lamarca, Engineering Smart Team Manager
F. Madden, Director, Regulatory Affairs
J. Mercer, Maintenance Rule Coordinator
J. Meyer, Technical Support Manager
W. Morrison, Maintenance Smart Team Manager
L. Pope, System Engineer
R. Segura, Nuclear Analyst Consultant (Electrical Systems)
R. Smith, Director, Operations
S. Smith, Director, System Engineering
D. Sparks, Senior Nuclear Analyst (Work Week Coordinator)
G. Struble, Operations Training Supervisor
J. Taylor, Engineering Smart Team Manager
C. Tran, Engineering Programs Manager
D. Wilder, Radiation and Industrial Safety Manager
H. Winn, System Engineer
G. Yezefski, System Engineer

NRC

D. Allen, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000445/2006004-01	NCV	Both Unit 1 Channels of Reactor Trip P4 Interlock Disabled in Mode 3 (Section 4OA2.4)

Closed

05000446/2005-003-00	LER	Two Pressurizer Safety Valves Found With Unsatisfactory Lift Setpoints (Section 4OA3)

Discussed

None

LIST OF DOCUMENTS REVIEWED

1R05: Fire Protection (71111.05Q)

FPI-103A, Unit 1 Safeguards Building Elevation 810'-6" Rad. Pen. Area & Elec. Equip Rooms, Revision 3

FPI-104A, Unit 1Train ‘A’ Diesel Generator & Equipment Elev.810' and Fuel Oil Day Tank Room Elev. 844', Revision 3

FIR-301, Portable Fire Extinguisher Inspection, Maintenance, Recharging, and Hydrostatic Testing, Revision 5

Comanche Peak Steam Electric Station Fire Protection Report, Unit 1 and Unit 2, Revision 25

1R12: Maintenance Effectiveness (71111.12)

Buildings and Structures

CPSES System Status -Buildings and Structures, 1ST Quarter FY06
CPSES System Status -Buildings and Structures, 2nd Quarter FY06

STA-744, “Maintenance Effectiveness Monitoring Program,” Revision 3
STA-744, Attachment 8.H, “Structural Monitoring Inspection Guide”
Form STA-744-2, “Structural Monitoring Area Inspection Form”
Form STA-744-3, “Structural Inspection Checklist”
Form STA-744-4, “Vertical Tank Inspection Checklist”
Form STA-744-5, “Component Supports Inspection Checklist”
Form STA-744-6, “Door Inspection Checklist”
Form STA-744-7, “Structural Monitoring Area Walkdown Form”

Nuclear Energy Institute “Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” (NUMARC 93-01, Revision 3)

Nuclear Energy Institute NEI 96-03, "Guidelines for Monitoring the Condition of Structures at Nuclear Power Plants"

Smart Forms:

SMF-2001-000410
SMF-2002-003058
SMF-2002-003610
SMF-2002-003690
SMF-2002-004239
SMF-2002-004288
SMF-2003-002531
SMF-2005-000206
SMF-2005-000682
SMF-2005-002066
SMF-2005-002569
SMF-2005-002652
SMF-2006-001335
SMF-2006-002024
SMF-2006-002404
SMF-2006-002507
SMF-2006-002561

<u>Diesel Fire Pumps X-05 and X-06</u>

System Health Report, Fire Protection, 2nd Quarter 2006
System Health Report, Fire Protection, 1st Quarter 2006

SMF-2005-000431-00
SMF-2006-002067-00
SMF-2006-002065-00
SMF-2006-002085-00
SMF-2006-002053-00
SMF-2006-002083-00
SMF-2005-003185-00
SMF-2006-001584-00
SMF-2006-002629-00

<u>1R15: Operability Evaluations (71111.15)</u>

<u>ARV Block Valves</u>

SMF-2003-000188
SMF-2006-002873

1R23: Temporary Plant Modifications (71111.23)

Final Design Authorization: FDA-2006-002529-01

Work Order: WO-4-06-169629

Smart Form: SMF-2006-002529

Course of Action: COA-2006-002529

2323-ES-100

Drawing 2323-E1-0005, Rev. 11, "480 V Auxiliaries One Line Diagram Safeguard Buses"

1EP2: Alert Notification System Testing (71114.02)

Comanche Peak Steam Electric Station Emergency Plan, Revision 33

Comanche Peak Alert and Notification System Design Report

CPSES letter CPSES-200601314 to Lisa Hammond, "Proposed changes to CPSES Alert and Notification System," July 11, 2006

2006 Granbury/Glen Rose Phone Book, "Yellow Book," 17^{th} edition, pages 4-9

Emergency Planning Staff Guidelines:

12, "Alert and Notification System Surveillance," Revision 11
16, "Emergency Preparedness Program Health," Revision 8
17, "Conducting Monthly Communication Equipment Checks," Revision 2

1EP3: Emergency Response Organization Augmentation Testing (71114.03)

Training Administrative Procedure TRA-105, "Emergency Preparedness Training," Revision 20

Emergency Planning Staff Guidelines:

15, "Remedial Training," Revision 3
22, "Emergency Preparedness Staff Training," Revision 0

Quarterly Call Out Drill Records, March 2005 through June 2006

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

SMART forms:

2004-2671	2004-3342	2005-0944	2005-3059	2006-1042
2004-2726	2004-3351	2005-1059	2005-3685	2006-1694
2004-2870	2004-3891	2005-1082	2005-4889	2006-1734
2004-2959	2004-3967	2005-1735	2005-5043-01	2006-1971
2004-3227	2005-0599	2005-2953	2006-0115	
2004-3304				

Nuclear Overview Department Audit Reports EVAL-2004-010, EVAL-2006-006

Self-Assessment Reports SA-2004-034, SA-2004-060, SA-2005-006, SA-2005-009, SA-2005-010, SA-2006-001

Station Administrative Procedures:

STA-424, "Self-assessment Program," Revision 0
STA-501, "Non-routine Reporting," Revision 12

Emergency Planning Staff Guideline 9, "Maintenance and Inventory of Equipment and Supplies," Revision 8

Emergency Plan Procedures:

100, "Maintaining Emergency Preparedness," Revision 6
121, "Re-entry, Recovery and Closeout," Revision 9
202, "Emergency Communications System and Equipment," Revision 7

Nuclear Training Procedure NTP-106, "Evaluation," Revision 11

4OA1: PI Verification (71151)

Emergency Exercise Reports; December 2005, April and May, 2006

Control Room Mini-drills; November and December 2005, July 2006

Emergency Plan Procedures:

201, "Assessment of Emergency Action Levels Emergency Classification and Plan Activation," Revision 11

304, "Protective Action Recommendations," Revision 18

Emergency Planning Staff Guideline 14, "Emergency Planning Master Schedule," Revision 3

Section 4OA2.2: Cumulative Effects of Operator Workarounds

Operator Work Around List - tracked by SMF 2003-1912, May 31, 2006
and SMF 2003-0460

Operator Compensatory Actions (LAN data base), printed 7/17/2006

Unit Differences (LAN database), printed 7/17/2006

Station Equipment Issues Second Quarter 2006 (part of Plan of the Day) July 13, 2006

Operational Focus Items (part of Plan of the Day) July 13, 2006

Course of Action (COA)

2004-1770-02-00
COA 2005-5005-01-00
COA 2006-0686-01-00
COA 2006-0987-01-00
COA 2006-1935-01-00
COA 2006-2030-01-00
COA 2006-2292-01-00

LIST OF ACRONYMS

ALARA	as low as reasonably achievable
CFR	*Code of Federal Regulations*
CPSES	Comanche Peak Steam Electric Station
EDG	emergency diesel generator
ERCOT	Electric Reliability Council of Texas
EVAL	evaluation
I&C	instrument and control
INC	instrument and control manual
LER	licensee event report
NCV	noncited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ODA	operations department administration
OPT	operations testing manual
PAR	protective action recommendation
PERC	plant event review committee
PI	performance indicator
SDP	significance determination process
SMF	smart form
SOP	system operating procedure
SSC	structures, systems, or components
TDAFWP	turbine driven auxiliary feed water pump
TGM	transmission grid management