



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

February 8, 2007

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 05000445/2006005 AND 05000446/2006005

Dear Mr. Blevins:

On December 31, 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 January 10, 2007, 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, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; 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 05000445/2006005 and 05000446/2006005
w/Attachment: Supplemental Information

cc w/Enclosure:
Fred W. Madden, Director
Regulatory Affairs
TXU Power
P.O. Box 1002
Glen Rose, TX 76043

George L. Edgar, Esq.
Morgan Lewis
1111 Pennsylvania Avenue, NW
Washington, DC 20004

Terry Parks, Chief Inspector
Texas Department of Licensing
and Regulation
Boiler Program
P.O. Box 12157
Austin, TX 78711

The Honorable Walter Maynard
Somervell County Judge
P.O. Box 851
Glen Rose, TX 76043

Richard A. Ratliff, Chief
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

Environmental and Natural
Resources Policy Director
Office of the Governor
P.O. Box 12428
Austin, TX 78711-3189

Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
Austin, TX 78711-3326

Susan M. Jablonski
Office of Permitting, Remediation
and Registration
Texas Commission on
Environmental Quality
MC-122
P.O. Box 13087
Austin, TX 78711-3087

Chairperson
Denton Field Office
Chemical and Nuclear Preparedness and Protection Division
Office of Infrastructure Protection
Preparedness Directorate
Dept. of Homeland Security
800 North Loop 288
Federal Regional Center
Denton, TX 76201-3698

Technological Services Branch
Chief
FEMA Region VI
800 North Loop 288
Federal Regional Center
Denton, Texas 76201-3698

Electronic distribution by RIV:
 Regional Administrator (**BSM1**)
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 DRS Deputy Director (**RJC1**)
 Senior Resident Inspector (**DBA**)
 Branch Chief, DRP/A (**CEJ1**)
 Senior Project Engineer, DRP/A (**TRF**)
 Team Leader, DRP/TSS (**MAS3**)
 RITS Coordinator (**MSH3**)

Only inspection reports to the following:

DRS STA (**DAP**)
 D. Cullison, OEDO RIV Coordinator (**DGC**)
ROPreports
 CP Site Secretary (**ESS**)

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: 05000445/2006005 and 05000446/2006005

Licensee: TXU Generation Company LP

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

Location: FM-56, Glen Rose, Texas

Dates: September 24, 2006 through December 31, 2006

Inspectors: D. Allen, Senior Resident Inspector
A. Sanchez, Resident Inspector
R. Azua, Reactor Inspector
B. Baca, Health Physicist
M. Haire, Resident Inspector (Temporary)
S. Rutenkroger, Regional Inspector

Approved by: Claude Johnson, Chief, Project Branch A
Division of Reactor Projects

Attachment: Supplemental Information

TABLE OF CONTENTS

SUMMARY OF FINDINGS	-3-
REPORT DETAILS	-4-
REACTOR SAFETY	-4-
1R04 <u>Equipment Alignment (71111.04)</u>	-4-
1R05 <u>Fire Protection (71111.05)</u>	-6-
1R08 <u>Inservice Inspection Activities (71111.08)</u>	-6-
1R11 <u>Licensed Operator Requalification Program (71111.11)</u>	-10-
1R12 <u>Maintenance Effectiveness (71111.12)</u>	-10-
1R13 <u>Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)</u>	-11-
1R15 <u>Operability Evaluations (71111.15)</u>	-12-
1R17 <u>Permanent Plant Modifications (71111.17A)</u>	-12-
1R19 <u>Postmaintenance Testing (71111.19)</u>	-13-
1R20 <u>Refueling and Outage Activities (71111.20)</u>	-14-
1R22 <u>Surveillance Testing (71111.22)</u>	-15-
1R23 <u>Temporary Plant Modifications (71111.23)</u>	-16-
RADIATION SAFETY	-16-
2OS1 <u>Access Control To Radiologically Significant Areas (71121.01)</u>	-16-
2OS2 <u>ALARA Planning and Controls (71121.02)</u>	-18-
OTHER ACTIVITIES	-20-
4OA1 <u>Performance Indicator Verification (71151)</u>	-20-
4OA2 <u>Problem Identification and Resolution (71152)</u>	-21-
4OA3 <u>Event Follow-up (71153)</u>	-22-
4OA5 <u>Other Activities</u>	-24-
4OA6 <u>Meetings, Including Exit</u>	-25-
SUPPLEMENTAL INFORMATION	A-1
KEY POINTS OF CONTACT	A-1
ITEMS OPENED, CLOSED, AND DISCUSSED	A-2
LIST OF DOCUMENTS REVIEWED	A-2
LIST OF ACRONYMS	A-9

SUMMARY OF FINDINGS

IR 05000445/2006005, 05000446/2006005; 09/24/2006-12/31/2006; Comanche Peak Steam Electric Station, Units 1 and 2. Access Control To Radiologically Significant Areas.

This report covered a 3-month period of inspection by two resident inspectors, three regional reactor inspectors and a health physicist. 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: Occupational Radiation Safety

- Green. The inspector reviewed a self-revealing noncited violation of 10 CFR 20.1902 for a failure to post a radiation area. The posting deficiency was identified during an investigation of a dosimeter dose alarm in Auxiliary Building Room 208. A radiological survey was performed two days prior with a radiation area being identified and documented on the survey, however, the radiation protection technician performing the survey failed to post the area. In addition, the lead technician who reviewed the survey failed to identify the posting deficiency. As an immediate corrective action, the licensee posted the area.

This finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affects the Occupational Radiation Safety cornerstone objective, in that the failure to post a radiation area could result in additional personnel exposure. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined that this finding was of very low safety significance because it did not involve: (1) an ALARA finding, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess doses. Additionally, this finding has a cross-cutting aspect in the area of human performance related to work practices because the radiation protection technicians failed to use error prevention tools such as self and peer checking to identify the posting deficiency. (Section 2OS1)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

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

Unit 2 began the reporting period at 100 percent power. The unit began power coastdown on September 27, 2006, and commenced a reactor shutdown on October 7 at 9:00 a.m. to begin refueling outage 2RF09. The reactor was manually tripped and entered Mode 3 at 12 noon that same day. On October 26 Unit 2 ended refueling outage 2RF09 when the main generator output breakers were closed at 3:57 a.m. On October 27 Unit 2 experienced a reactor trip due to HI-HI steam generator level signal from Steam Generator 2-02. Later that same day, the Unit 2 main generator output breakers were closed at 5:17 p.m. On October 29 the reactor was manually tripped from 80 percent power due to the failure of Flow Control Valve 2- FCV-530, which led to the rapid lowering of Steam Generator 2-03 level. On October 31 the Unit 2 main generator output breakers were closed at 4:22 a.m. The unit achieved 100 percent power on November 2 at 10:00 a.m. and remained at that power for the rest of the reporting period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Partial system Walkdown (71111.04Q)

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 Diesel Generator 2-01 in accordance with System Operating Procedure (SOP) Manual SOP-609B, "Diesel Generator System," Revision 9, while Component Cooling Water Pump 2-02 was inoperable due to planned oil drain and flush on November 2, 2006
- Unit 1 Motor Driven Auxiliary Feedwater Pump 1-01 in accordance with Operations Testing Manual (OPT) Procedure OPT-206A, "AFW System," Revision 25, and SOP-304A, "Auxiliary Feedwater System," Revision 16, while Motor Driven Auxiliary Feedwater Pump 1-02 was inoperable for scheduled surveillance testing on November 2, 2006

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

.2 Detailed Semiannual System Walkdown (71111.04S)

a. Inspection Scope

The inspectors conducted a detailed inspection of Units 1 and 2 feedwater systems, primarily focusing upon the feedwater control valves and feedwater control bypass valves and supporting systems to verify the functional capability of the system as described in the Final Safety Analysis Report. During the walkdowns, inspectors examined system components for correct alignment, for electrical power and instrument air availability, and for material conditions of structural components that could degrade system performance. In addition, the inspectors referenced and used the following documents to verify proper system alignment and setpoints:

- Final Safety Analysis Report, Chapter 10.4.7, "Condensate and Feedwater Systems," Amendment No. 100b
- CPSES Drawing M1-2203, "Instrumentation & Control Diagram Steam Generator Feed Water System CHAN 0510/0540, 2130/2133, 2158/2165," Revision CP-15
- CPSES Drawing M1-0203, "Flow Diagram Steam Generator Feedwater System," Revision CP-24
- Copes-Vulcan Drawing No. E-333079, "12 inch Class 900 Valve Assembly - 16 inch Ends," Revision 4

The inspectors also reviewed recent corrective action documents, recent work requests, temporary modifications, and design issues to determine if any of these items could affect the system's ability to perform as designed. The inspectors interviewed appropriate plant staff regarding the system's maintenance history. A field walkdown was completed during the week of December 17, 2006.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

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 2CA101 - Unit 2 containment on October 16, 2006
- Fire Zone 2SE018 - Unit 2 Train B switchgear room on October 17, 2006
- Fire Zone 2SE016 - Unit 2 safeguards building 832' elevation electrical equipment room on October 18, 2006
- Fire Zone 2SB004 - Unit 2 safeguards building 790' elevation corridor on October 18, 2006
- Fire Zone SE018 - Unit 1 Train B switchgear room on October 19, 2006
- Fire Zone TB110 - Unit 1 non-safety related switchgear room on October 19, 2006

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

Inspection Procedure 71111.08 requires four samples, as identified in Sections 02.01, 02.02, 02.03, and 02.04.

.1 Performance of Nondestructive Examination Activities Other Than Steam Generator Tube Inspections, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The inspection procedure requires the review of nondestructive examination activities consisting of two or three different types (i.e., volumetric, surface, or visual). The inspectors observed the performance of ultrasonic examinations (volumetric) on two of the Unit 2 pressurizer spray line welds and two containment spray line welds for Valves 2-HV-4758 and 2-HV-4759. Plus, the inspectors observed penetrant examinations (surface) on the two containment spray line welds for Valves 2-HV-4758 and 2-HV-4759. The inspectors also reviewed radiographic examinations (volumetric) of four containment spray line welds. In addition, the inspectors observed four visual (VT-3) examinations performed on component supports, and a containment spray line weld as well. The table below identifies the above examinations which were conducted using four methods and three different examination types.

System/ Component	Identity	Examination Type	Examination Method
Pressurizer Spray	Pipe to Elbow Weld	Volumetric	Ultrasonic
Pressurizer Spray	Pipe to Elbow Weld	Volumetric	Ultrasonic
Containment Spray	Pipe to Valve 2-HV-4758 Welds	Volumetric	Ultrasonic Radiography
Containment Spray	Pipe to Valve 2-HV-4758 Welds	Surface	Penetrant
Containment Spray	Pipe to Valve 2-HV-4759 Welds	Volumetric	Ultrasonic Radiography
Containment Spray	Pipe to Valve 2-HV-4759 Welds	Surface	Penetrant
Safety Injection Component Support	Vertical Snubber H3: SI-2-089-403-C41K	Visual	Visual (VT-3)
Safety Injection Component Support	Horizontal Snubber H6: SI-2-089-404-C41K	Visual	Visual (VT-3)
Safety Injection Component Support	Vertical Snubber H5: SI-2-089-405-C41K	Visual	Visual (VT-3)
Safety Injection Component Support	Large Bore Pipe Support H1: SI-2-089-402-C41K	Visual	Visual (VT-3)
Reactor Vessel	Vessel Flange	Visual	Visual (VT-1)

For each of the observed nondestructive examination activities, the inspectors verified that the examinations were performed in accordance with the specific site procedures and the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements.

During review of each examination, the inspectors verified that appropriate nondestructive examination procedures were used, examinations and conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also verified the nondestructive examination certifications of the personnel who performed the above volumetric, surface, and visual examinations. Finally, the inspectors observed that indications identified during the ultrasonic, radiographic, and visual examinations were dispositioned in accordance with the ASME qualified nondestructive examination procedures used to perform the examinations.

The inspection procedure requires review of one or two examinations with recordable indications that were accepted for continued service to ensure that the disposition was made in accordance with the ASME Code. The inspectors were informed that no indications exceeding ASME Code allowables were known to be in service.

The inspection procedure further requires verification of one to three welds on Class 1 or 2 pressure boundary piping to ensure that the welding process and welding examinations were performed in accordance with the ASME Code. The inspectors verified through record review that welding performed on a containment spray system isolation valve, both in the shop and in the field, was performed in accordance with Sections IX and XI of the 1995 Edition of the ASME Code. This included review of welding material issue slips to establish that the appropriate welding materials had been used and verification that the welding procedure specification had been properly qualified.

The inspectors completed the one sample required by Section 02.01.

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The inspection requirements for this section parallel the inspection requirement steps in Section 02.01. However, the inspectors were informed that no Reactor Vessel Upper Head Penetration Activities were scheduled to be performed during this refueling outage. Thus, this inspection sample was not performed.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection Activities (Pressurized Water Reactors)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of those systems that could be deleteriously affected by boric acid corrosion.

The inspection procedure requires review of a sample of boric acid corrosion control walkdown visual examination activities through either direct observation or record review. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown, as specified in Station Administrative Manual (STA) Procedure STA-737, "Boric Acid Corrosion Detection and Evaluation," Revision 4. Samples of documented visual inspection records of inspection walkdowns performed on components and equipment during the previous Refueling Outage 2RF08, and this refueling outage, were reviewed by the inspectors.

Additionally, the inspectors performed independent observations of piping containing boric acid during walkdowns of the containment building and the auxiliary building.

The inspection procedure requires verification that visual inspections emphasize locations where boric acid leaks can cause degradation of safety significant components. The inspectors verified through direct observation and program/record review that the licensee's boric acid corrosion control inspection efforts are directed towards locations where boric acid leaks can cause degradation of safety-related components.

The inspection procedure requires both a review of one to three engineering evaluations performed for boric acid leaks found on reactor coolant system piping and components, and one to three corrective actions performed for identified boric acid leaks. There were no applicable Smart Forms generated since the last inspection period that required formal engineering evaluations, (e.g., that resulted in a separate design or structural engineering analysis to determine continued operability). The inspectors reviewed Smart Forms documenting minor valve packing leaks on valves in the safety injection system. The planned corrective actions were adequate in each case.

The inspectors completed the one sample required by Section 02.03.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure specified performance of an assessment of in situ screening criteria to assure consistency between assumed nondestructive examination flaw sizing accuracy and data from the Electric Power Research Institute examination technique

specification sheets. It further specified assessment of appropriateness of tubes selected for in situ pressure testing, observation of in situ pressure testing, and review of in situ pressure test results. However, the inspectors were informed that no "Steam Generator Tube Inspection Activities," were scheduled for this outage. Thus, this inspection sample was not performed

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 inspectors observed two licensed operator requalification exam scenarios in the control room simulator on December 12, 2006. The first scenario included a failure of a main turbine control valve, failure of a pressurizer pressure channel with a stuck open pressurizer power operated relief valve, Control Rod Bank B continuous withdrawal, and concluded with a steam generator tube rupture.

The second scenario included a failure of the main generator voltage regulator, followed by two dropped rods, an anticipated transient without a trip, and concluded with a faulted steam generator.

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 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 attempted transition of the circulating water system from Maintenance Rule a(1) status to a(2) status, which was postponed due to the expert panel requiring a more thorough explanation of the circulating water motors and pumps failure histories, corrective actions taken, and a review of the performance criteria.
- The Unit 2 Feedwater Control Valve 2-FCV-530 functional failure that caused a manual reactor trip on October 29, 2006, and was documented in the corrective action program as Smart form SMF-2006-003660-00.

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 Unit 2RF09 Outage Risk Assessment and defense-in-depth contingency plans on October 2-6, 2006
- The scheduling of emergent work on 345kV Switchyard Breaker 8090 for an air leak and the opening of 345kV Switchyard Breaker 8040 for transmission line work offsite on November 17, 2006
- Rescheduling of Unit 2 Train B emergency diesel generator post-24 hour run re-torque of the injectors due to inclement weather, followed by performing Unit 2 turbine driven auxiliary feedwater pump governor valve inspection and surveillance test on December 1, 2006
- The issuance of an Emergency Electric Curtailment Plan-Step 1, by the Electric Reliability Council of Texas, due to a moderate grid disturbance on December 22, 2006

The inspectors completed four 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 SDP 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:

- SMF-2006-002787-00, setting discrepancies for Siprotec Multi-Function Relays for the emergency diesel generators between vendor software configuration Document 38-5038773-04, relay setting Calculation EE-CA-0008-0267 and Design Basis Document DBD-EE-051, and the field configuration as documented in Work Order WO-3-02-318811-01, reviewed on November 21, 2006
- SMF-2006-002795-00, degraded bus voltage time delay relays having inconsistent descriptions between the Technical Specification Bases and the Final Safety Analysis Report, reviewed on November 22, 2006
- SMF-2006-003083-00, degraded heat trace for the north vent stack Wide Range Gas Monitor, reviewed on November 22, 2006

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A)

a. Inspection Scope

For the following permanent plant modification described below, the inspectors reviewed the Final Design Authorizations (FDA) FDA-2005-003364-02-01, 02, 03, and 04, the

10 CFR 50.59 screening, implementing work orders, installation and post-installation testing procedures, and observed installation and testing of portions of the modification to verify that design bases, license bases, and performance capability had not been degraded through this modification.

- The replacement of the Unit 2 containment spray pumps suction valves from the refueling water storage tank, Valves 2-HV-4758 and 4759, in conjunction with the containment recirculation sump modification. This modification consisted of the replacement of motor operated gate valves with faster acting motor operated butterfly valves. This would allow more refueling water storage tank volume to be pumped to containment before transfer of pump suctions to the containment sump. This provided the higher containment sump water level necessary for net positive suction head for emergency core cooling system recirculation following a loss of coolant accident. This modification affected only Unit 2 and did not require a license amendment.

The inspectors completed one sample.

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 A Motor Driven Auxiliary Feedwater Pump Discharge Control Valves 2-PV-2453A and 2-PV-2453B, following preventative maintenance and auxiliary feedwater accumulator check valve leak rate testing on Valves 2AF-0291, 2AF-0236, 2AF-0237, and 2AF-0238 in accordance with OPT-601B, "TRN A MDAFW Accumulator Check Valve Leak Test," Revision 4, completed on October 10, 2006
- Unit 2 Train A and B reactor trip breakers and reactor trip bypass breakers following preventative maintenance, in accordance with MSE-P0-6342, "Reactor Trip Switchgear Inspection and Maintenance," Revision 6, on October 12-14, 2006
- Unit 2 Main Feedwater Control Valves 2-FCV-0530, 2-FCV-0540 and Feedwater Bypass Valves 2-LV-2164 and 2-LV-2165 following installation of Herion solenoids on the feedwater control valves, and installation of new ASCO solenoids on the feedwater bypass valves, and tested in accordance with WO-4-06-171177-00, WO-4-06-171191-00, WO-2-06-171198-00, and WO-2-06-171196-00 on October 30, 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 three samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated licensee's 2RF09 activities to ensure that risk was considered when developing and when deviating from the outage schedule, the plant configuration was controlled in consideration of facility risk, mitigation strategies were properly implemented, and Technical Specification requirements were implemented to maintain the appropriate defense-in-depth. Specific outage inspections performed and outage activities reviewed and/or observed by the inspectors included:

- Discussions and review of the outage schedule concerning risk with the Outage Manager
- Unit shutdown and cooldown
- Containment walkdowns to identify indications of reactor coolant leakage, evaluate material condition of equipment not normally available for inspection, inspect fire protection equipment and fire hazards, observe radiation protection postings and barriers, and evaluate coatings and debris for potential impact on the recirculation containment sumps
- Reduced inventory activities to perform vacuum fill of reactor coolant system
- Reactor coolant system instrumentation including Mansell level instrumentation
- Defense in depth and mitigation strategy implementation
- Containment closure capability
- Verification of decay heat removal system capability
- Spent fuel pool cooling capability
- Reactor water inventory control including flow paths, configurations, alternate means for inventory addition, and controls to prevent inventory loss
- Controls over activities that could affect reactivity

- Refueling activities that included fuel offloading, fuel transfer, and core reloading
- Implementation of procedures for foreign material exclusion
- Electrical power source arrangement
- Containment cleanup and inspection
- Containment recirculation sump inspection after modification of sump filters
- Alloy 600 inspections of pressurizer top and bottom penetration welds
- Unit heatup and startup
- Licensee identification and resolution of problems related to refueling activities

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 2 ECCS check valve operability test in accordance with OPT-521B, "ECCS Operability," Revision 3, observed on October 19, 2006
- Unit 2 Residual Heat Removal Pump 2-01 operability test in accordance with OPT-203B, "Residual Heat Removal System," Revision 11, reviewed on December 15, 2006

The inspectors completed two 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 modifications were 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 2 Feedwater Bypass Valves 2-LV-2164 and 2-LV-2165 being modified from utilizing Herion solenoids to ASCO solenoids according to TM 02-06-000006, FDA-2006-003660-01-01, and work orders WO-2-06-171198-00 and WO-2-06-171196-00, observed and reviewed on October 30-31, 2006.
- Unit 1 Train B safety injection pump lube oil cooler flow indication being temporarily obtained by installing a portable ultrasonic flow meter according to WO 4-06-169947-00 and evaluated in EVAL-2006-002785-01-00, reviewed on November 20, 2006.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control To Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspector used the requirements in 10 CFR Part 20, the Technical Specifications, and the licensee's procedures required by Technical Specifications as criteria for determining compliance. During the inspection, the inspector interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspector performed independent radiation dose rate measurements and reviewed the following items:

- Controls (surveys, posting, and barricades) of radiation, high radiation, and airborne radioactivity areas in the reactor, fuel, and auxiliary buildings
- Radiation work permits, procedures, engineering controls, and air sampler locations
- Conformity of electronic personal dosimeter alarm setpoints with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in one airborne radioactivity area
- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Radiation work permit briefings and worker instructions
- Adequacy of radiological controls such as, required surveys, radiation protection job coverage, and contamination controls during job performance
- Dosimetry placement in high radiation work areas with significant dose rate gradients
- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas
- Controls for special areas that have the potential to become very high radiation areas during certain plant operations
- Posting and locking of entrances to accessible high dose rate - high radiation areas and very high radiation areas
- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone
- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

The inspector completed 21 of the required 21 samples.

b. Findings

Introduction. The inspector reviewed a self-revealing noncited violation of 10 CFR 20.1902 for a failure to post a radiation area. The posting deficiency was identified during an investigation of a dosimeter dose alarm in Auxiliary Building Room 208. The violation had very low safety significance (Green).

Description. On October 12, 2006, during an investigation of a dosimeter dose alarm in Auxiliary Building Room 208, the licensee identified a radiation area posting deficiency. A radiological survey was performed two days prior with a radiation area being identified and documented on the survey, however, the radiation protection technician performing the survey failed to post the area. In addition, the lead technician who reviewed the survey failed to identify the posting deficiency.

Analysis. The failure to post a radiation area is a performance deficiency. The finding is greater than minor because it is associated with the occupational radiation safety exposure control attribute and affected the cornerstone objective to provide adequate safety to workers from unintended exposure to radiation. The failure to post a radiation area created the potential for increased individual doses over a two day period and was contrary to regulations. Because this occurrence involved potential workers' unplanned, unintended dose contrary to regulations, this finding was evaluated with the Occupational Radiation Safety SDP. The finding was determined to be of very low safety significance (GREEN) because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. Additionally, this finding has a cross-cutting aspect in the area of human performance related to work practices because the radiation protection technicians failed to use error prevention tools such as self and peer checking to identify the posting deficiency.

Enforcement. In part, 10 CFR 20.1902 states that radiation areas shall be posted with a conspicuous sign bearing the radiation symbol and the words, "Radiation Area." Contrary to regulations, the surveyed radiation area was not posted and presented an area for unintended increase of worker exposure for two days. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (Smart Form 2006-3331), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000445;446/2006005-01, Failure to Post a Radiation Area.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The

inspector used the requirements in 10 CFR Part 20 and the licensee's procedures required by technical specifications as criteria for determining compliance. The inspector interviewed licensee personnel and reviewed:

- Site specific ALARA procedures
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents
- Dose rate reduction activities in work planning
- Post-job (work activity) reviews
- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered
- Exposure tracking system
- Workers use of the low dose waiting areas
- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner
- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas
- Self-assessments, audits, and special reports related to the ALARA program since the last inspection
- Resolution through the corrective action process of problems identified through post-job reviews and post-outage ALARA report critiques
- Corrective action documents related to the ALARA program and follow-up activities such as initial problem identification, characterization, and tracking
- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

The inspector completed 6 of the required 15 samples and 9 of the optional samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

.1 Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

The inspector reviewed licensee documents from April 1, 2006, through September 30, 2006. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee's technical specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in Nuclear Energy Institute (NEI) document 99.02). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

The inspector completed the required sample (1) in this cornerstone.

.2 Public Radiation Safety Cornerstone

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual
Radiological Effluent Occurrences

The inspector reviewed licensee documents from April 1, through September 30, 2006. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

The inspector completed the required sample (1) in this cornerstone.

.3 Mitigating Systems Cornerstone

Safety System Unavailability Indicators were not inspected during Calendar Year 2006, in accordance with TI 2515/169, "Mitigating Systems Performance Index Verification," (see Section 4OA5.2).

b. Findings

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 follow-up, 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 meetings.

b. Findings

No findings of significance were identified.

.2 Semiannual Trend Review

a. Inspection Scope

On December 30, 2006, the inspectors completed a semiannual review of licensee internal documents, reports, and performance indicators to identify trends that might indicate the existence of more safety significant issues. The inspectors reviewed the following types of documents:

- Corrective Action Documents (Smart Forms)
- System Health Reports
- Planned Maintenance Work Week Critiques
- CPSES Nuclear Overview Department Evaluation Reports (Audits)
- Human Performance Program Health Indicators Package
- Corrective Action Program Health report
- CPSES Self-Assessment Reports

b. Findings and Observations

No findings of significance were identified. However, during the review, the inspectors did note trends or concerns that had been identified by the licensee and/or NRC which warrant continued attention. These included (1) foreign material exclusion, (2) human performance, specifically procedural violations, and (3) equipment issues, specifically quality spare components. The inspectors did not identify any additional trends.

The inspectors determined that the licensee had adequately identified adverse trends and entered them into the corrective action program using an appropriate threshold.

.3 Radiation Safety Inspection

a. Inspection Scope

The inspector evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 Unit 2 Reactor Trip Due to a Secondary Transient Initiated During Load Reject Testing

a. Inspection Scope

On October 27, 2006, CPSES was operating at 28 percent power and performing the third 25 MWe load swing test as part of the Operational Acceptance Testing for the main turbine digital controls upgrade when a secondary transient developed. The transient resulted in a Steam Generator 2-02 HI-HI level, which tripped the main feedwater pump. Operators initiated a manual reactor trip at 3:08 a.m. Inspectors responded to the site and reported to the control room. The inspectors discussed the trip event with operations, engineering and licensee management to gain an understanding of the event and to access followup actions. The inspectors also reviewed operator logs, procedure use, computer printouts, and walked down the control boards. The licensee's posttrip review package was reviewed in accordance with the procedure Operations Department Administration Manual ODA-108, "Post RPS/ESF Actuation Evaluation," Revision 14.

The Operational Acceptance Testing included the intentional introduction of transients to evaluate the response of various plant control systems, their interactions with each other, and their effects on plant operation. The transients were initiated by introducing a rapid turbine generator load reduction. The test procedure contained precautions and limitations to ensure plant parameters were monitored and actions would be taken to prevent the plant from operating outside of analyzed safe conditions. These actions were consistent with normal operating practices, including taking manual control of equipment when automatic control was not stable, and tripping the turbine and/or reactor if an automatic trip setpoint is approached or exceeded.

Prior to the third load swing, operators had raised Tave by withdrawing control rods 12 steps, and increased the feedwater pump suction pressure by placing a second condensate pump in service and aligning a heater drain pump to the suction of the main feedwater pump. These actions were taken to increase reactor coolant temperature to offset the effects of Xenon build up in the core.

When the third load swing was initiated, oscillations in the steam flow, most likely the result of the steam dump valves cycling near their full closed position, caused the

feedwater control system to also oscillate. The feedwater pump speed demand, which responds to changes in steam pressure and feedwater pressure, began cycling in a divergent manner. When the operators placed feedwater pump speed control in manual, the controller output was apparently at a peak, causing high feedwater pressure and increasing levels in the steam generators. With the steam and feedwater flow still oscillating, the operators began placing the feedwater flow control valves in manual, but the level in Steam Generator 2-02 reached the Hi-Hi setpoint before the operators could terminate the level increase.

The licensee has determined the root cause to be the initiation of a secondary transient that the main feedwater, heater drain, and steam dump control systems could not dampen. There were several differences between the first two tests and the third test, which consisted of: 1) forward flow established and in automatic to maintain Tave-Tref deviation at zero for an RCS leak rate test just prior to the third test, 2) a slightly higher reactor coolant system Tave, and 3) an increasing main steam line header pressure, as opposed to a decreasing header pressure.

The Root Cause Analysis performed by the licensee identified several corrective actions to avoid repeating this plant trip. Procedure guidance for sequencing secondary system pumps will be provided to ensure the main feedwater pump steam control valve remains in an effective throttling range. Engineering will evaluate dampening for control inputs to secondary system controls. Gain adjustments made to the main feedwater pump speed controller prior to the outage were changed back to the previous settings. Training will be provided on the lessons learned from this plant trip.

b. Findings

No findings of significance were identified.

.2 Unit 2 Reactor Trip Due to Feedwater Regulating Valve Malfunction

a. Inspection Scope

On October 29, 2006 at 3:20 p.m., while Unit 2 was at 80 percent power and holding for xenon stabilization, a manual reactor trip was initiated due to Steam Generator 2-03 level lowering uncontrollably. The inspectors responded to the site and reported to the control room. The inspectors discussed the trip event with operations, engineering and licensee management to gain an understanding of the event and to assess follow-up actions. The inspectors also reviewed operator logs, procedure use, computer printouts, and walked down the control boards. The licensee's posttrip review package was reviewed in accordance with the procedure Operations Department Administration Manual ODA-108, "Post RPS/ESF Actuation Evaluation," Revision 14.

The licensee has determined the root cause to be a loose wire on Solenoid Valve SV-2 associated with Feedwater Regulating Control Valve 2-FCV-530. The loose wire resulted in the loss of air between the valve positioner and the valve operator diaphragm, which caused the flow control valve to fail in the closed position. The licensee was able to recreate and prove the failure in their testing laboratory.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Implementation of Temporary Instruction (TI) 2515/166 - Pressurized Water Reactor Containment Sump Blockage

a. Inspection Scope

The objective of this TI is to support the NRC review of licensees' activities in response to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors (PWRs)." This TI requires NRC inspectors to verify actions implemented in response to NRC Generic Letter 2004-02 are complete and, where applicable, are programmatically controlled. It is not the objective of this TI to determine the adequacy of the licensee actions taken in response to Generic Letter 2004-02. The Office of Nuclear Reactor Regulation will review licensee Generic Letter responses and conduct audits to assess the adequacy of licensee actions.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Performance Index Verification (Temporary Instruction 2515/169)

This TI provided the guidelines to verify that the licensee has correctly implemented the Mitigating Systems Performance Index (MSPI) guidance for reporting unavailability and unreliability of the monitored systems. The safety systems that CPSES is required to monitor are: Emergency Alternating Current (EAC), High Pressure Safety Injection (HPSI), Auxiliary Feedwater (AFW), Residual Heat Removal (RHR), Station Service Water (SSW), and Component Cooling Water (CCW).

a. Inspection Scope

The inspector validated the unavailability and unreliability input data and verified the accuracy of reporting results for the second quarter of 2006. Specifically, the inspector:

- Interviewed the MSPI reporter
- Reviewed implementing procedures
- Observed data collection and documentation, and a demonstration of the input of the data into consolidated data entry
- Reviewed the list of surveillance activities which, when performed, do not render the train unavailable due to the short duration of the activity (less than 15 minutes) and surveillance activities where operator action is credited for availability

- Reviewed the MSPI basis document and the 2002-2004 input data in accordance with NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 4
- Independently confirmed, on a sampling basis, the accuracy of the actual and planned unavailability, and the reliability data for the monitored components

b. Findings

No findings of significance were identified. The inspector concluded that the licensee is monitoring, collecting and entering the appropriate data in accordance to the prescribed guidance. The inspector has provided the following details of the inspection as required by TI 2515/169.

1. Did the licensee accurately document the baseline planned unavailability hours for the MSPI systems?

The licensee has accurately documented the baseline planned unavailability hours for the MSPI systems. The inspector did identified a very minor issue regarding the exclusion of 1.15 hours of planned unavailability. The licensee has entered the issue into the corrective action program as SMF-2006-004204-00.

2. Did the licensee accurately document the actual unavailability hours for the MSPI systems?

The licensee has accurately documented the actual unavailability hours for the MSIP systems.

3. Did the licensee accurately document the actual unreliability information for each MSPI monitored component?

The licensee has accurately documented the actual unreliability information for each MSPI monitored component.

4. Did the inspector identify significant errors in the reported data, which resulted in a change to the indicated index color?

No significant errors in the reported data were identified.

5. Did the inspector identify significant discrepancies in the basis document which resulted in (1) a change to the system boundary; (2) an addition of a monitored component; or (3) a change in the reported index color?

No significant discrepancies in the basis document were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 18, 2006, the engineering inspectors presented the results of the inservice inspection review to Mr. R. Flores, Site Vice President, and other members of licensee management. Licensee management acknowledged the inspection findings.

On October 20, 2006, the inspector presented the occupational radiation safety inspection results to Mr. M. Kanavos, Plant Manager, and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On January 10, 2007, the inspectors presented the resident inspection results to Mr. R. Flores, Site Vice President, 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
S. Bradley, Supervisor, Health Physics, Radiation Protection & Safety Services
T. Clouser, Manager, Shift Operations
J. Curtis, Radiation Protection Manager, Radiation and Industrial Safety
R. Flores, Site Vice President
J. Gallman, Senior Nuclear Analyst (Work Week Coordinator)
B. Henley, Engineering Consultant (Seismic Analysis)
D. Holland, Senior Nuclear Analyst (Work Week Coordinator)
T. Hope, Manager, Regulatory Performance
M. Kanavos, Plant Manager
S. Karpyak, Risk & Reliability Engineering Supervisor
R. Kidwell, Sr. Nuclear Technologist, Regulatory Affairs
G. Krishnan, Procurement Engineering & Program Manager, SHAW
D. Kross, Director, Maintenance
J. Lamarca, Engineering Smart Team Manager
F. Madden, Director, Regulatory Affairs
S. Maier, Design Engineering Analysis Manager, Technical Support
J. Mercer, Maintenance Rule Coordinator
J. Meyer, Technical Support Manager
W. Morrison, Maintenance Smart Team Manager
D. O'Connor, Supervisor, Radiation Protection, Radiation Protection & Safety Services
P. Passalugo, ISI Engineer, SHAW Engineering Programs
L. Pope, System Engineer
J. Seawright, Consulting Engineer, Regulatory Affairs
R. Segura, Nuclear Analyst Consultant (Electrical Systems)
R. Smith, Director, Operations
S. Smith, Director, System Engineering
D. Sparks, Senior Nuclear Analyst (Work Week Coordinator)
J. Taylor, Engineering Smart Team Manager
C. Tran, Engineering Programs Manager
I. Whitt, Engineer, Boric Acid Corrosion Detection Program
D. Wilder, Radiation and Industrial Safety Manager
H. Winn, System Engineer
G. Yezefski, System Engineer

NRC

D. Allen, Senior Resident Inspector
A. Sanchez, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000445, 446/2006005-01

NCV

Failure to Post a Radiation Area.
(Section 2OS1)

Closed

None

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R05: Fire Protection (71111.05Q)

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

FPI-102B, Unit 2 Safeguards Building Elevation 790'-0", Revision 2

FPI-106B; Unit 2 Safeguards Building, Elevation 831'-6" Corridor, RB Assess, Elect. Equip. Area, Revision 3

FPI-107A, U1 Safeguards Building, Elevation 852'-6" Electrical Equipment Area & FW Penetration Area, Revision 3

FPI-107B, U2 Safeguards Elevation. 852' Electrical Equipment Area & Feedwater Penetration Area, Revision 2

FPI-201B, Unit 2 Containment Building Elev. 808'-0", Revision 1

FPI-202B, Unit 2 Containment Building Elev. 832'-6", Revision 1

FPI-203B, Unit 2 Containment Bldg. Elevation 860'-0", Revision 1

FPI-204B, Unit 2 Containment Building, Elev. 905'-0", Revision 1

FPI-304A; Unit 1, Switchgear, Central Alarm, and HVAC Equipment Rooms, Revision 3

SMF-2006-003520-00

SMF-2006-003747-00

Drawing M2-1900-SG-03, Penetration Seal Map, Revision CP2 and CP3

Drawing SG-810-083-8, Penetration Seal Map RM. 83, Safeguard - Unit 1, Revision CP-1

DCA 75340, Revision 0

Section 1R08: Inservice Inspection Activities

Procedures

WLD-103 Welder Performance Qualifications, Revision 6

WLD-101 Welding Program Requirements

WLD-105 Welding Material Storage and Control

WLD-106 ASME/ANSI General Welding Requirements

STA-737 Boric Acid Corrosion Detection and Evaluation, Revision 4

WCI-607 Fluid Leak Management Process, Revision 1

TX-ISI-302 Ultrasonic Examination of Austenitic Piping Welds, Revision 2

TX-ISI-88 Underwater Remote Visual Examination of Reactor Vessel and Internals for CPSES, Revision 3

TX-ISI-11 Liquid Penetrant Examination for Comanche Peak Steam Electric Station, Revision 11

TX-ISI-8 VT-1 and VT-3 Examination Procedure for CPSES, Revision 6

RT-1 ACUREN NDE Procedure - Radiographic Examination, Revision 10

Drawings

BRP-CT-2-SB-032 Containment Spray, Sheet 1 of 2, Revision CP-7

BRP-CT-2-SB-033 Containment Spray, Sheet 1 of 2, Revision CP-7

SI-2-090-402-C41S Safety Injection System, Revision CP-5

TCX-4202 Safety Injection Large Bore Pipe Support, Revision 4

Smart Forms

SMF-2006-001092-00	SMF-2006-001768-00	SMF-2006-000147-00
SMF-2006-004124-00	SMF-2005-004027-01	SMF-2003-000838-01
SMF-2004-002056-01	SMF-2005-004312-00	SMF-2005-004629-00
SMF-2006-000648-00	SMF-2006-001302-00	SMF-2006-002591-00

Work Orders

WO 2-04-157004-00	WO 2-04-157005-00	WO 2-04-157006-00
WO 4-05-160932-00	WO 4-05-160806-00	WO 4-03-149225-00
WO 3-00-334691-01	WO 2-04-158327-00	WO 2-02-143484-00
WO 3-05-343942-01		

Miscellaneous

EVAL-2005-000945-02-00 "Results of Walk Downs and Inspections for Boric Acid Leaks and/or Corrosion Performed for 2RF08"

TXX-06129 "Inservice Inspection Plan for Unit 2 Refueling Outage No. 9"

Section 1R17: Permanent Plant Modifications (71111.17A)

Final Design Authorizations (FDA)

2005-3364-02-04

Smartforms

2006-3288
2006-3277

Work Orders

2-06-166419-00
2-06-166420-00
2-06-167062-00
2-06-167063-00
2-06-167913-00
2-06-166976-00
2-06-166977-00
2-06-169025-00
2-06-169027-00
4-06-166425-00
4-06-166424-00

Miscellaneous

VL-06-002507
Drawing BRP-CT-2-SB-032
Drawing BRP-CT-2-SB-033

Section 1R19: Postmaintenance Testing (71111.19)

WO-3-05-344087-01
WO-3-05-344086-01
WO-3-05-327011-01
WO-3-05-327013-01

WO-3-05-327010-01
WO-3-05-327012-01

Section 2OS1: Assess Controls to Radiologically Significant Areas (71121.01)

Audits and Self-Assessments

Nuclear Overview Surveillances dated: 05/15/06, 05/19/06, 10/06/06
Quality Assurance Surveillance dated: 10/11/06
SA-2006-036

General Assess Permits

2006-01, 2006-21

Procedures

RPI-110	Radiation Protection Shift Activities, Revision 12
RPI-402	Personnel Decontamination, Revision 16
RPI-602	Radiological Surveillance and Posting, Revision 29
RPI-614	Skin Dose Calculations, Revision 4
RPI-622	Containment Refueling Job Coverage, Revision 1
STA-650	General Health Physics Plan, Revision 5
STA-653	Contamination Control Program, Revision 9
STA-656	Radiation Work Control, Revision 12
STA-660	Control of High Radiation Areas, Revision 9

Radiation Work Permits

2006-2100, 2006-2406, 2006-2500, 2006-2600, 2006-2602, 2006-2603

Smart Forms

2006-3222, 2006-3330, 2006-3331, 2006-3417, 2006-3455, 2006-3469

Miscellaneous

Locked High Radiation Area Log
Radiation Survey Records

Section 2OS2: ALARA Planning and Controls (71121.02)

Audits and Self-Assessments

SA-2006-042, SA-2006-048
Nuclear Overview Surveillances dated: 07/30/06, 08/31/06, 10/06/06

Radiation Work Permits

2006-2100, 2006-2406, 2006-2500, 2006-2600, 2006-2602, 2006-2603

Procedures

RPI-606 Radiation Work and General Assess Permits, Revision 14
STA-650 General Health Physics Plan, Revision 5
STA-651 ALARA Program, Revision 9
STA-656 Radiation Work Control, Revision 12

Smart Forms

2006-1644, 2006-1821, 2006-2448, 2006-3402, 2006-3404, 2006-3416

Section 40A1: Performance Indicator Verification (71151)

Procedures

Radiation Safety NRC Performance Indicators - Job Aide, Definition, and Flow Chart, 02/14/06

Smart Forms

2006-2756, 2006-3225

Section 40A2: Problem Identification and Resolution (71152)

Human Performance (Procedural Adherence)

Smart Forms

2006-2387	2006-3576
2006-2403	2006-3583
2006-3122	2006-3600
2006-3179	2006-3724
2006-3242	2006-3974
2006-3269	

Equipment Issues (Quality of Parts)

Smart Forms

2006-2171	2006-3005
2006-2184	2006-3664
2006-2169	2006-3939
2006-3159	2006-4188
2006-2155	2006-2884
2006-2175	2006-3083
2006-2182	2006-1766
2006-2231	2006-3253

Smart Forms

COE	Audit	2005-4643	2006-2439
2006-2498	2006-2505	2005-4727	2006-3078
2006-2577	2006-2941	2006-3176	
2006-2936	2006-2974	2006-3127	
2006-3119	2006-3014	2006-3762	
2006-3140		2006-3577	
		2006-4064	

Section 40A3: Event Follow-up

Smart Forms

2006-2632
2006-3660

Section 40A5: Other Activities

TI 2515/166, Pressurized Water Reactor Containment Sump Blockage

CPSES-200501776 Comanche Peak Steam Electric Station Response to Requested Information Part 2 of NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 1, 2005

TI 2515/169, Mitigating Systems Performance Index Verification

1st Quarter 2006 MSPI Data
2nd Quarter 2006 MSPI Data
2005 Actual Unavailability for Unit 1 (Complete)
2005 Actual Unavailability for Unit 2 (Sampled)
2004 1st Quarter data for Unit 1 AFW
2004 3rd Quarter data for Unit 1 EDG
2004 4th Quarter data for Unit 1 RHR
2004 2nd Quarter data for Unit 2 SSW
2004 3rd Quarter data for Unit 2 SSW, CCW
Archived LCO Data for March 2005

Procedures

OPT-216A, "Remote Shutdown Operability Test," Revision 10
OPT-463B, "Train A Safeguards Slave Relay K601 Actuation Test," Revision 7
MDA-1105, "Maintenance Department Data Sheet Program," Revision 3

Smart Form

2002-2566
2002-3813
2003-0158
2006-1677
2006-4202
2006-4204

Preventative Maintenance
PM-306600

NRC Regulatory Issue Summary 2006-07: Changes to the Safety System Unavailability
Performance Indicators, Date June 12, 2006

Mitigating Systems Performance Indicator Desktop, Revision 4
MSPI Deviation Report-Unreliability 2003-2006: Unit 1 Emergency AC System
MSPI Deviation Report-Unavailability 2003-2006: Unit 1 Emergency AC System
Consolidate Data Entry FAQ 289-297

LIST OF ACRONYMS

2RF09	unit 2, ninth refueling outage
ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
CFR	<i>Code of Federal Regulations</i>
CPSES	Comanche Peak Steam Electric Station
FDA	final design authorization
MSPI	mitigating systems performance index
NCV	noncited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OPT	operations testing manual
SDP	significance determination process
SMF	smart form
SOP	system operating procedure
SSC	structures, systems, or components
STA	station administrative manual
TI	temporary instruction
WO	work order