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

July 26, 2004

EA-04-133

Mr. M. R. Blevins, Senior Vice President
and Principal Nuclear Officer
TXU Energy
ATTN: Regulatory Affairs
Comanche Peak Steam Electric Station
P.O. Box 1002
Glen Rose, Texas 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000445/2004003 AND 05000446/2004003 AND NOTICE OF VIOLATION

Dear Mr. Blevins:

On June 23, 2004, 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 June 24, 2004, with Mr. J. J. Kelley 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. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of the inspection, the NRC has determined that a Severity Level IV violation of NRC requirements occurred. This violation was evaluated in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is included on the NRC's Web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Enforcement Policy**. The violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it are described in detail in the enclosed inspection report. The violation is being cited in the Notice because it was determined to have been committed willfully and plant personnel failed to promptly provide information concerning the violation to appropriate NRC personnel, in accordance with Section VI.A.1.d(1) of the Enforcement Policy. Therefore, the finding could not be treated as a noncited violation (NCV).

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence is already adequately documented on the docket in the enclosed inspection report. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In this case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

The enclosed report also documents one NRC-identified finding and two self-revealing findings of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these three findings as NCVs consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis of 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, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the 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/

William D. Johnson, Chief
Project Branch A
Division of Reactor Projects

Dockets: 50-445
50-446
Licenses: NPF-87
NPF-89

Enclosures:
1. Notice of Violation
2. NRC Inspection Report 05000445/2004003 and 05000446/2004003
w/attachment: Supplemental Information

cc w/enclosures:
Fred W. Madden
Regulatory Affairs Manager
TXU Generation Company LP
P.O. Box 1002

Glen Rose, Texas 76043

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

G. R. Bynog, Program Manager/
Chief Inspector
Texas Department of Licensing & Regulation
Boiler Division
P.O. Box 12157, Capitol Station
Austin, Texas 78711

County Judge
P.O. Box 851
Glen Rose, Texas 76043

Chief, Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, Texas 78756-3189

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

Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
1701 North Congress Avenue
Austin, Texas 78701-3326

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

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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Branch Chief, DRP/A **(WDJ)**
Senior Project Engineer, DRP/A **(TRF)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(KEG)**
Brian McDermott **(BJM)**
CP Site Secretary **(ESS)**
W. A. Maier, RSLO **(WAM)**
NRR/DIPM/EPB/EPHP **(EWW)**
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Send Section 4OA5 (1.) of IR to NRR/DE/EMCB, Attention: Ted Sullivan or **(EJS)**
G. F. Sanborn, D:ACES **(GFS)**
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7/13/04	7/13/04	7/13/04	7/14/04	7/14/04	7/20/04

C:DRS/PEB	C:DRP/A				
LJSmith	WDJohnson				
/RA/	***/RA/***				
7/13/04	7/26/04				

NOTICE OF VIOLATION

TXU Energy | Docket Numbers 50-445; 50-446

Comanche Peak Steam Electric Station | License Numbers NPF-87; NPF-89
EA-04-133

During an NRC inspection conducted May 24-28, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> Technical Specification 5.7.1.e requires, in part, that high radiation areas (greater than 0.1 rem per hour to 1 rem per hour at 30 centimeters from the source of radiation) be accessed only after dose rates in the area have been determined and entry personnel are knowledgeable of them.
>
> Contrary to the above, on November 4, 2003, a contract worker accessed a high radiation area in Room 1-061 of the Unit 1 safeguards building and was not knowledgeable of dose rates in the area. Dose rates within the room were as high as 250 millirems per hour at 30 centimeters from the source of radiation.
>
> This is a Severity Level IV violation (Supplement IV). (05000445; 446/2004003-01)

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation to prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in Inspection Report No. 50-445/2004003; 50-446/2004003. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation," include the EA number, and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. Therefore, to the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

In accordance with 10 CFR 19.11, you are required to post this Notice within two working days.

Dated this 26 day of July 2004

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-445, 50-446
Licenses:	NPF-87, NPF-89
Report:	05000445/2004003 and 05000446/2004003
Licensee:	TXU Generation Company LP
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56, Glen Rose, Texas
Dates:	March 25 through June 23, 2004
Inspectors:	D. B. Allen, Senior Resident Inspector A. A. Sanchez, Resident Inspector D. R. Carter, Health Physicist C. E. Johnson, Senior Reactor Inspector M. E. Murphy, Senior Operations Engineer L. T. Ricketson, P.E., Senior Health Physicist W. C. Sifre, Reactor Inspector J. I. Tapia, Senior Reactor Inspector
Accompanying Personnel:	P. J. Elkmann, Emergency Preparedness Analyst B. W. Tindell, Reactor Inspector, Division of Reactor Safety V. M. Klein, Nuclear Safety Intern Office of Nuclear Reactor Regulation
Approved by:	W. D. Johnson, Chief, Project Branch A Division of Reactor Projects
Attachment:	Supplemental Information

SUMMARY OF FINDINGS

Comanche Peak Steam Electric Station, Units 1 and 2
NRC Inspection Report 05000445/2004003, 05000446/2004003

IR 05000445/2004003, 05000446/2004003; 03/25/2004-06/23/2004; Comanche Peak Steam Electric Station, Units 1 & 2; Access Control to Radiologically Significant Areas.

This report covered a 3-month period of inspection by resident inspectors and announced inspections by regional reactor and health physics inspectors. One Severity Level IV violation and three Green noncited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process". Findings for which the Significance Determination Process does not apply may be Green or may 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

- SL IV. On November 4, 2003, an individual entered a high radiation area without contacting radiation protection personnel for a briefing on the dose rates in the area, despite verbal and posted instructions to the contrary. Dose rates within the room were as high as 250 millirems per hour at 30 centimeters from the source of radiation. The licensee was alerted to the situation when the individual's electronic dosimeter alarmed because the dose rate setpoint was exceeded. The occurrence was a violation of Technical Specification 5.7.1.e., and the circumstances surrounding it indicate it was committed willfully. The licensee identified this Severity Level IV violation and entered it into the corrective action program. However, it did not meet all of the criteria in Section VI.A.1 of the NRC Enforcement Policy for treatment as a Non-Cited Violation because the licensee did not promptly provide information concerning the violation to NRC personnel. Therefore, the violation is being documented in a Notice of Violation.

  The failure to contact radiation protection personnel for a briefing on radiation dose rates prior to entering a high radiation area is a performance deficiency because it resulted in the licensee's failure to meet a requirement in its technical specifications. Because there are willful aspects of the violation, it is subject to traditional enforcement. The willful aspects notwithstanding, the inspector used the Occupational Radiation Safety Significance Determination Process described in Manual Chapter 0609, Appendix C, to analyze the significance of the finding. The inspector determined that the finding was of very low safety significance 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. The finding was entered into the licensee's corrective action program as

SMF-2003-3594, and the individual was appropriately disciplined. This finding also had crosscutting aspects associated with human performance (Section 2OS1).

- Green. The NRC identified two examples of a noncited violation of 10 CFR 20.1501a because the licensee failed to perform surveys to identify dose rates and contamination levels of potential radiological hazards. On January 8, 2004, workers performing decontamination of a pole that was used for filter compaction alarmed the contamination monitors while exiting the radiologically controlled area. The licensee identified that the pole had contact dose rates of 150 millirem per hour; however, the inspector determined that the pole was not surveyed for contamination. In addition, on April 5, 2004, the inspector identified dose rates as high as 250 millirem per hour on contact and 80 millirem per hour at 30 centimeters on a containment spray line in Piping Area X-213. The posted survey map outside the room indicated general area dose rates near the pipe of between 1 and 5 millirem per hour.

  The failure to perform surveys to evaluate the magnitude and extent of radiation levels and the concentrations or quantities of radioactive materials is a performance deficiency. The finding is greater than minor because it is associated with the Occupational Radiation Safety cornerstone attribute of Program and Process and affected the cornerstone objective to ensure adequate protection of a worker's health and safety from exposure to radiation. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because it was not associated with as low as is reasonably achievable issues, there was no overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. The two examples of the finding were entered into the licensee's corrective action program as SMF-2004-1264 and SMF-2004-0069 (Section 2OS1).

- Green. The inspector reviewed a self-revealing noncited violation of Technical Specification 5.4.1 for failure to follow a radiation work permit requirement. On April 4, 2004, scaffold builders constructed scaffolding up into an area of containment that had not been surveyed by radiation protection personnel and received an electronic dosimeter dose rate alarm.

  The failure to follow radiation work permit requirements is a performance deficiency. The finding was greater than minor because it was associated with the Occupational Radiation Safety cornerstone attribute of Program and Process and affected the cornerstone objective to ensure adequate protection of a worker's health and safety from exposure to radiation. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because the finding was not associated with as low as is reasonably achievable issues, there was no overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. The finding was entered into the licensee's corrective action program as SMF-2004-1202 (Section 2OS1).

- Green. The NRC reviewed two examples of a self-revealing noncited violation of Technical Specification 5.7.1e for the failure of personnel to receive a briefing on radiation dose rates prior to entering a high radiation area. On February 10, 2004, an individual entered the Waste Monitor Tank Room X-185, a posted high radiation area, without being briefed on dose rates in the area and received an electronic dosimeter dose rate alarm. On February 18, 2004, an individual entered the piping penetration Train A, Room 077B, a posted high radiation area, without being briefed on the dose rates in the area before being stopped by another worker.

  The failure to be briefed about radiation dose rates prior to entering a high radiation area is a performance deficiency. The finding was greater than minor because it was associated with the Occupational Radiation Safety cornerstone attribute of Program and Process and affected the cornerstone objective to ensure adequate protection of a worker's health and safety from exposure to radiation. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because the finding was not associated with as low as is reasonably achievable issues, there was no overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. The two examples of the finding were entered into the licensee's corrective action program as SMF-2004-0620 and SMF-2004-0471 (Section 2OS1).

B. Licensee Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

CONTENTS

REPORT DETAILS

Summary of Plant Status

Comanche Peak Steam Electric Station (CPSES) Unit 1 began the period at essentially 100 percent power. On March 27, 2004, the unit was shut down for the tenth Refueling Outage (1RF10). The outage ended on May 4, 2004, at 7:22 a.m. when the generator output breakers were closed. The unit achieved approximately 100 percent power on May 11, 2004. The unit operated at essentially full power for the remainder of the report period.

CPSES Unit 2 operated at essentially 100 percent power for the entire report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

On May 11, 2004, the inspectors observed the control room operators implement Abnormal Operating Procedure (ABN) ABN-907, "Acts of Nature," Revision 10, in response to a lightning storm and associated partial loss of power to the 25kV loop. The inspectors evaluated whether the operators adequately addressed the actions specified in the procedure and whether the procedure specified the actions that should be taken to protect safety-related equipment during severe weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors conducted partial walkdowns of the following three risk-significant systems to verify that they were in their proper standby alignment as defined by system operating procedures and system drawings. During the walkdowns, inspectors examined system components for materiel conditions that could degrade system performance. In addition, the inspectors evaluated the effectiveness of the licensee's problem identification and resolution program in resolving issues which could increase event initiation frequency or impact mitigating system availability.

- Unit 1 Train A safety injection system in accordance with System Operating Procedure (SOP) SOP-201A, "Safety Injection System," Revision 13, while the Train B safety injection system was inoperable due to scheduled lube oil cooler cleaning, on May 17, 2004

- Unit 1 Train A containment spray system in accordance with SOP-204A, "Containment Spray System," Revision 13 and Operations Testing Procedure (OPT) OPT-205A, "Containment Spray System," Revision 13, while the Train B containment spray system was inoperable due to scheduled surveillance testing, on June 15, 2004

- Unit 1 Train B safety chilled water system in accordance with SOP-815A, "Safety Chilled Water System," Revision 12, and OPT-209A, "Safety Chilled Water System," Revision 10, while the Train A safety chilled water system was inoperable due to scheduled maintenance on June 22, 2004

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors assessed the licensee's control of transient combustible materials, the materiel condition and lineup of fire detection and suppression systems, and the materiel condition of manual fire equipment and passive fire barriers during tours of the following seven risk-significant areas. The licensee's fire preplans and Fire Hazards Analysis Report were used to identify important plant equipment, fire loading, detection and suppression equipment locations, and planned actions to respond to a fire in each of the plant areas selected. Compensatory measures for degraded equipment were evaluated for effectiveness.

- Fire Zone 1CA101 - Unit 1 containment building on April 5, 2004

- Fire Zone EC051 - Unit 1 Train B inverter room on April 23, 2004

- Fire Zone EC053 - Unit 1 Train A inverter room on April 23, 2004

- Fire Zone EC050 - Unit 2 Train B inverter room on April 23, 2004

- Fire Zone EC052 - Unit 2 Train A inverter room on April 23, 2004

- Fire Zone 1SE16 - Unit 1 safeguards building electrical equipment area on April 23, 2004

- Fire Zone 2SD009 - Unit 2 Train A switchgear room on April 23, 2004, and May 3, 2004

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors conducted an inspection of flood protection measures implemented at Comanche Peak from June 9 through June 11, 2004, due to heavy rainfall in the area. This included a review of the Design Basis Document DBD-CS-071, "Probable Maximum Flood (PMF)," Revision 71, to verify that the assumptions made in the external flooding analysis remained valid. The review also included the Technical Requirements 13.7.34 and ABN-907, "Acts of Nature," Revision 10, to ensure that all appropriate actions were taken by the operations staff for the Squaw Creek Reservoir level greater than 777.5 feet.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors observed the eddy-current testing and inspection of one of the risk significant heat exchangers cooled by the station service water system, the Unit 1 Train A component cooling water heat exchanger during the week of April 5, 2004. The inspectors also reviewed the station service water system fouling program and test data results, as appropriate, for the following systems:

- Unit 1 component cooling water heat exchangers the week of April 5, 2004

- Unit 1 and Unit 2 emergency diesel generator (EDG) jacket water heat exchangers on June 9, 2004

The review also included the technical specifications, component and system health reports, smart forms (SMF), calculations, and an interview with the system engineer.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections

a. Inspection Scope

If the licensee completed welding on the pressure boundary for Class 1 or 2 systems since the beginning of the previous outage, the procedure requires verification that acceptance and preservice examinations were done in accordance with the American Society of Mechanical Engineers (ASME) Code for one to three welds. There was no welding since the beginning of the previous outage for the inspectors to review for this inspection.

The procedure also requires verification that one or two ASME Code Section XI repairs or replacements meet code requirements. The inspectors reviewed six records of ASME Code Section XI repairs or replacement activities to verify that the repair or replacement activities were in accordance with Section XI requirements.

The inspectors also reviewed the relief request submitted on February 15, 2001, for the application of risk-informed inservice inspection for Class 1 and 2 piping. This request was approved by the NRC in a letter dated September 28, 2001.

b. Findings

No findings of significance were identified.

.2 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 (EPRI) examination technique specification sheets. It further specified assessment of appropriateness of tubes selected for in-situ pressure testing, the inspection procedure specified observation of in-situ pressure testing, and review of in-situ pressure test results. At the time of this inspection there were no pressure tests performed, however, the inspector reviewed the licensee’s procedures for pressure testing.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess the licensee’s prediction capability. The inspectors reviewed the licensee’s report, "Steam Generator Degradation Assessment for 1RF10 Outage," dated March 24, 2004. The purpose of the assessment is to identify degradation mechanisms and for each mechanism to determine proper detection technique, determine number of tubes, establish structural limits, and establish flaw growth rates.

The inspection procedure specified confirmation that the steam generator tube eddy-current test scope and expansion criteria meet technical specification requirements, EPRI guidelines, and commitments made to the NRC. The inspector reviewed the steam generator tube eddy-current test scope and expansion criteria.

The inspection procedure required confirmation that the licensee inspected all areas of potential degradation, especially areas which were known to represent potential eddy-current test challenges (e.g., top-of-tubesheet, tube support plates, and U-bends). The inspector confirmed that all known areas of potential degradation, including eddy-current test-challenged areas, were included in the scope of inspection and were being inspected.

The inspection procedure further required verification that repair processes being used were approved in the technical specifications for use at the site. The inspectors reviewed the licensee's procedure for tube sleeving and plugging. The inspector also verified a sample of sleeve candidate tubes and verified that they satisfied the sleeving criteria.

The inspection procedure also required confirmation of adherence to the technical specification plugging limit. The inspection procedure required determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspector confirmed that the licensee was adhering to these specifications.

If steam generator leakage greater than 3 gallons per day was identified during operations or during post-shutdown visual inspections of the tubesheet face, the inspection procedure required verification that the licensee had identified a reasonable cause and corrective actions for the leakage based on inspection results. The inspector determined that leakage greater than 3 gallons per day did not exist.

The inspection procedure required confirmation that the eddy-current test probes and equipment were qualified for the expected types of tube degradation and assessment of the site specific qualification of one or more techniques . The inspector observed portions of all eddy-current tests performed. During these examinations, the inspectors verified that (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered to, and (4) probe travel speed was in accordance with procedural requirements.

Finally, the inspection procedure specified review of one to five samples of eddy-current test data if questions arose regarding the adequacy of eddy-current test data analyses. The inspector did not identify any results where eddy-current test data analyses adequacy was questionable.

b. <u>Findings</u>

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed selected inservice inspection related SMFs issued during the current and past refueling outages. The review served to verify that the licensee's corrective action process was being correctly utilized to identify conditions adverse to quality and that those conditions were being adequately evaluated, corrected and trended. The inspectors determined that the licensee's threshold for initiating SMFs was low, thereby capturing virtually all deficiencies identified in the inservice inspection program. The inspectors also concluded that corrective actions were being appropriately addressed.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Biennial Inspection

a. Inspection Scope

The inspector performed an in-office review of the annual operating examination test results for 2003. Since this was the first half of the biennial requalification cycle, the licensee had not yet administered the written examination. These results were assessed to determine if they were consistent with NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," Revision 8, Supplement 1, guidance and Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process," requirements. This review included examination of test results, which included no failures, for a total of 87 licensed operators.

b. Findings

No findings of significance were identified

.2 Quarterly Licensed Operator Requalification Activities Review

a. Inspection Scope

The inspectors observed a licensed operator training session in the control room simulator on June 8, 2004. The simulator sets were focused on immediate operator actions per Emergency Response Guidelines EOP 0.0A, "Reactor Trip Or Safety Injection," Revision 7. The inspectors also observed a staff simulator session on June 17, 2004. This scenario included: a digital rod position indication half accuracy, feed water tube leak, a load reduction, power range malfunction, steam leak inside containment and loss of offsite electrical power, and a loss of heat sink. 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 attended and reviewed classroom sessions concerning: design modifications, current industry events, immediate operator actions, and human performance during the week of June 7, 2004.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (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 two equipment performance problems:

- The Unit 2 EDG 2-01 failure and replacement of one tachometer, which led to the diesel being inoperable on May 25, 2004. This issue was placed into the corrective action program as SMF-2004-002076-00.

- The Unit 2 Instrument Air Compressors 2-01 and 2-02 series of trips associated with operator and clearance errors. These issues were placed into the corrective action program as SMF-2003-001378-00, SMF-2003-003839-00, and SMF-2004-002081-00.

The inspectors reviewed whether the structures, systems, or components (SSCs) 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.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed five 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:

- 1RF10 midloop operation with fuel in the reactor vessel in order to remove steam generator manways and install nozzle dams on March 30 and 31, 2004

- Entry of the increase in the Unit 2 EDG 2-01 fail to start probability in the Safety Monitor database, which is used for on-line risk evaluations, according to a Probabilistic Risk Assessment Evaluation, "Risk Impacts Associated with Missed Surveillance of the Loss of Power Diesel Generator Start Instrumentation," Revision 1, on April 2, 2004

- Emergent work to troubleshoot and repair EDG 2-01 while Train A component cooling water heat exchanger was drained for planned heat exchanger annual cleaning on May 25, 2004

- Unit 2 EDG 2-01 surveillance run and entry into procedure ABN-907, "Acts of Nature," Revision 10, in response to a severe thunderstorm watch on June 3, 2004

- Emergent repair of hydraulic leak on Main Steam Isolation Valve 1-02 with EDG 1-01 inoperable due to planned maintenance and surveillance testing on June 23, 2004

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Evolutions and Events (71111.14)

a. Inspection Scope

For the nonroutine event described below, the inspectors reviewed operator logs, procedure use, plant computer data, and applicable SMFs and interviewed operators to determine what occurred and to determine if the operator response was in accordance with plant procedures. When applicable the inspectors also attended Plant Event Review Committee meetings.

- On March 30, 2004, the reactor operators lowered Unit 1 reactor coolant water level to 57 inches above the reactor core (Midloop) in preparation for removal of the steam generator manways and installation of nozzle dams. The inspectors reviewed Generic Letter No. 88-17, "Loss of Decay Heat Removal," and TXU's responses to the Generic Letter. Integrated Plant Operating Procedure IPO-010A, "Reactor Coolant System Reduced Inventory Operations," Revision 14, was reviewed to ensure adequate controls were in place. The inspectors verified the required instrumentation and equipment alignments were established. The control room activities and operators' actions were observed during the evolution to ensure the procedure was followed, the plant parameters were closely monitored, conservative decisions were made, and that the evolution was completed safely.

- On April 3, 2004, Unit 2 Reactor Coolant Pump Seal Injection Filter 2-02 Drain Valve 2CS-8386B failed full open when a plant equipment operator attempted to close the valve more tightly to terminate leakage past the seat. Control room operators responded to indications of loss of reactor coolant pump seal injection, vibration alarm on Reactor Coolant Pump 2-01, decreasing volume control tank level and decreasing pressurizer level. The operators reduced letdown flow by isolating the 45 gallon per minute letdown orifice, entered ABN-101, "Reactor Coolant Pump Trip/Malfunction," Revision 9, placed in service Seal Injection Filter 2-01 and isolated Seal Injection Filter 2-02, which terminated the loss of coolant. Initial metallurgical analysis of the valve indicated that the cause of the failure was stress corrosion cracking in the handwheel bushing. In addition to the attributes above, the inspectors reviewed the past failure history of this and similar valves, and the safety significance of the event.

- On April 18, 2004, the reactor operators entered Mode 6 and commenced core reload. The inspectors observed the control room activities and operators' actions to ensure proper procedure usage, the plant parameters were closely monitored, communications between operations and the fuel handling personnel, core reactivity and fuel location tracking, and technical specification adherence.

- On April 27, 2004, the reactor operators again lowered the Unit 1 reactor coolant water level to midloop in preparation for removal of the nozzle dams and reinstallation of the steam generator manways. The inspectors again verified the required plant conditions were in place and observed the control room activities to ensure the evolution was performed safely.

- On June 11, 2004, the Unit 2 reactor operators reduced reactor power to 75 percent in order to conduct a surveillance activity on the turbine protective devices and the turbine control and stop valves. The control room activities and operators' actions were observed during the evolution to ensure the appropriate procedures were used and followed, plant parameters were closely monitored, conservative decisions were made, and that the evolution was completed safely.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors selected five operability evaluations conducted by CPSES personnel involving risk-significant systems or components. The inspectors evaluated the technical adequacy of the licensee's operability determination, determined whether appropriate compensatory measures were implemented, and determined whether or not other pre-existing conditions were considered, as applicable. Additionally, the inspectors evaluated the adequacy of the CPSES problem identification and resolution program as it applied to operability evaluations. Specific operability evaluations reviewed are listed below:

- Evaluation EVAL-2004-001811-01, determine operability of 1-TE-0433A, Unit 1 Reactor Coolant System (RCS) Hot Leg 1-03 wide range temperature element, found to be outside the allowable range during RTD cross calibration testing at RCS temperature of 550 degrees, reviewed on May 4, 2004

- Quick Technical Evaluation (QTE) QTE-2004-001881-01-00, determine operability of containment electrical penetration backup protection breakers, found to not meet the design base document DBD-EE-062 requirements for penetration conductor protection, reviewed on May 10, 2004

- QTE-2004-001650-01-00, determine operability of the Unit 1 EDG 1-01 due to the discovery of the control room Var meter not reading correctly during acceptance testing for the voltage regulator modification, reviewed on June 9, 2004

- QTE-2004-001177-00, determine the operability of seven time delay relays in the Unit 2 Train A EDG loss of power diesel generator start instrumentation, which were not tested according to Technical Specifications, reviewed on June 21, 2004

- QTE-2004-002308-01-01, determine operability of Wide Range Gaseous Monitors X-RE-5570A and X-RE-5570B with associated heat tracing circuits below the 85 percent of full current acceptance criteria, reviewed on June 23, 2004

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 eleven maintenance activities:

- Unit 1 Station Service Water Discharge Check Valve 1-02 disassembly and maintenance in accordance with OPT-207A, "Service Water System," Revision 12, on April 10, 2004

- Unit 1 EDG 1-02 outage maintenance in accordance with work order (WO) WO-3-02-336976-01 and procedure Maintenance Section-Electrical Section MSE-PO-0861, "Startup and Break-In Run for Emergency Diesel Generator with DSC Digital Governors," Revision 3, on April 11, 2004

- Unit 1 EDG 1-02 following a replacement of the Phase C linear reactor and associated silicon control rectifiers in accordance with WO-4-04-154478-00 and Maintenance Section-Electrical Section procedure MSE-P0-0861, "Startup and Break-In Run for Emergency Diesel Generator with DSC Digital Governors," Revision 3, on April 13, 2004

- Unit 1 Train A Component Cooling Water Valves 1-HV-4572-MO and 1-HV-4574-MO outage maintenance in accordance with CPSES testing manual, procedure PPT-P1-6200, "CCW to RHR/CS outlet Valve Flow Control Test," Revision 2, on April 19, 2004

- Unit 1 Station Service Water 1-01 motor replacement in accordance with OPT-207A, "Service Water System," Revision 12, on April 19, 2004

- Unit 1 Centrifugal Charging Pump 1-01 outboard bearing housing replacement and motor alignment in accordance with OPT-201A, "Charging System," Revision 13, on April 20, 2004

- Unit 1 Train A safety injection sequencer following the replacement of the 1-K103-A relay and the contact cartridge for the 1-K615-A relay in accordance with WO-4-04-154767-00 and WO-4-04-154768-00, respectively, and OPT-430A, "Train A Integrated Test Sequence," Revision 3, on April 25-26, 2004

- Unit 1 Train A reactor trip switchgear undervoltage and shunt trip test switch pushbutton replacement and reactor trip bypass breaker inspection and maintenance in accordance with OPT-443A, "Reactor Trip Breaker and Stationary Gripper Coil Response Time," Revision 3, on April 29, 2004

- Unit 1 Steam Generator 1-01 feedwater isolation valve repair of a body to bonnet leak and outage maintenance according to OPT-511A, "FW Section XI Isolation Valves," Revision 11, on April 30, 2004

- Unit 1 turbine driven auxiliary feedwater pump following outage maintenance and Woodward governor replacement in accordance with OPT-206A, "AFW System," Revision 24, on May 2, 2004

- Unit 2 EDG 2-01 following the replacement of one of the Channel II tachometer, its associated power supply and Relay 2-HX/3419-9 in accordance with WO-4-04-155464-00, and SOP-609B, "Diesel Generator System," Revision 7, on May 25, 2004

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 determine if the testing was adequate to verify equipment operability.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated licensee's 1RF10 activities to ensure that risk was considered 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 activities reviewed and/or observed by the inspectors include:

- Discussions and review of the outage schedule concerning risk with the Outage Manager
- Reduced inventory and midloop activities to perform steam generator nozzle dam removal and manway installation
- Verified RCS 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
- Electrical power source arrangement
- Containment cleanup and inspection
- Containment recirculation sump inspection
- 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; and the effectiveness of the licensee's problem identification and correction program. The following nine surveillance test activities were observed and/or reviewed by the inspectors:

- Unit 1 Train B Integrated Test Sequence in accordance with OPT-435A, "Train B Integrated Test Sequence," Revision 3, section 8.2, on April 13, 2004

- Unit 1 emergency core cooling system check valves, in accordance with OPT-521A, "ECCS Check Valve Operability," Revision 1, on April 21, 2004

- Unit 2 EDG 2-02 in accordance with OPT-214B, "Diesel Generator Operability Test," Revision 12, on April 22, 2004

- Unit 1 reactor vessel head and pressurizer vent path verification in accordance with OPT-505A, "Reactor Coolant Valve Operability Test," Revision 7, on April 28, 2004

- Unit 1 low power physics testing in accordance with Nuclear Engineering Procedure NUC-301, "Low Power Physics Testing,"Revision 10, on May 4, 2004

- Unit 1 control rod drop time test in accordance with Nuclear Engineering Procedure NUC -206, "Control Rod Drop Timing (Plant Computer Method)," Revision 14, on May 4, 2004

- Unit 1 RCS resistance temperature detectors and thermocouples calibration in accordance with Instrumentation and Control Procedure INC-7919A, "Reactor Coolant System RTD Cross Calibration," Revision 1, on May 4, 2004

- Unit 1 Loop 1-02 N16 power monitor in accordance with Instrumentation and Control Procedure INC-7662A, "Channel Calibration N16 Power Monitor Module," Revision 3, on May 11, 2004

- Unit 2 turbine overspeed protection system, in accordance with OPT-217B, "Turbine Overspeed Protection System Test," Revision 7, on June 12, 2004

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following temporary modification and associated documentation. The temporary modification was verified to be installed and administratively controlled in accordance with plant documentation and procedures.

- Installation and testing of the alternate power generators in accordance with SOP-614A, "Alternate Power Generator Operation," Revision 7, to support refueling outage 1RF10, reviewed on March 29, 2004

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

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 three radiation, high radiation, or airborne radioactivity areas

- Radiation work permit, procedure, and engineering controls and air sampler locations

- Conformity of electronic personal dosimeter alarm set points 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 two airborne radioactivity areas
- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) 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 (No licensee events reports or special reports were identified)
- Corrective action documents related to access controls
- Radiation work permit briefings and worker instructions (Reactor vessel under-head inspection)
- 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 (Reactor vessel under-head inspection and steam generator nozzle dam installation)
- Controls for special areas that have the potential to become very high radiation areas during certain plant operations
- Posting and locking of entrances to all 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:

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas
- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies
- Adequacy of the licensee’s internal dose assessment for any actual internal exposure greater than 50 millirem CEDE
- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone

Therefore, the inspector completed 21 of the required 21 samples.

b. Findings

1. Introduction: The inspector reviewed a Severity Level IV violation of Technical Specification 5.7.1.e that resulted from an individual's failure to obtain a briefing on radiation dose rates prior to entering a high radiation area.

Description: On November 4, 2003, a contract worker entered a high radiation area without contacting radiation protection personnel for a briefing on the dose rates in the area. The individual was looking for trash and entered Room 1-061 of the Unit 1 safeguards building. The room was posted with signs which read, "High radiation area, not routinely surveyed, contact RP prior to entry." Dose rates within the room were as high as 250 millirems per hour at 30 centimeters from the source of radiation. In addition, licensee representatives stated that the worker was told not to enter high radiation areas. This item is considered to be self-revealing because the licensee was alerted to the situation when the individual's electronic dosimeter alarmed after exceeding its dose rate setpoint.

Analysis: The individual's failure to contact radiation protection personnel for a briefing on radiation dose rates prior to entering a high radiation area is a performance deficiency because it resulted in the licensee's failure to meet a requirement in its technical specifications. Because the circumstances surrounding the violation indicate it was committed willfully, it is subject to traditional enforcement. The willful aspects notwithstanding, the inspector used the Occupational Radiation Safety Significance Determination Process described in Manual Chapter 0609, Appendix C, to analyze the significance of the finding. The inspector determined that the finding was of very low safety significance 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.

Enforcement: Technical Specification 5.7.1.e requires that high radiation areas (greater than 0.1 rem per hour to 1 rem per hour at 30 centimeters) be accessed only after dose rates in the area have been determined and entry personnel are knowledgeable of them. The individual violated this requirement on November 4, 2003, by entering a high radiation without contacting radiation protection personnel and obtaining a briefing on the dose rates in the area. The circumstances surrounding the violation indicate it was committed willfully. The finding was entered into the licensee's corrective action program as SMF-2003-3594, and appropriate disciplinary action was taken against the individual involved. The violation also involved the actions of a low-level individual. However, it did not meet all of the criteria in Section VI.A.1 of the NRC Enforcement Policy for treatment as a Non-Cited Violation because the licensee did not promptly provide information concerning the violation to NRC personnel. Therefore, the violation is being documented in a Notice of Violation: VIO 05000445; 446/2004003-01, Entry into a high radiation area without a briefing on radiation dose rates (EA-04-133).

2. Introduction. The NRC identified two examples of a Green NCV of 10 CFR 20.1501(a) because the licensee failed to perform radiological surveys to identify dose rates and contamination levels of potential radiological hazards.

Description. On April 5, 2004, during a walkdown of the auxiliary building Piping Area X-213, the inspectors identified elevated dose rates on piping at head height just inside the room. There were no radiological signs or warnings of the higher dose rates. The survey map located outside the room indicated general area dose rates of between 1.5 to 5 millirem per hour. The licensee performed a survey of the area and identified on contact dose rates of as high as 270 millirem per hour and general area dose rates of as high as 80 millirem per hour.

During a review of Corrective Action Document SMF-2004-0069, the licensee reported that on January 7, 2004, a pole used to distribute RCS filters within a high integrity container was bagged and moved to a decontamination booth in the fuel building. The health physics technician performed a radiation survey of the pole and identified dose rates of 150 millirem per hour on contact and 15 millirem per hour at 30 centimeters. However, the technician failed to perform a contamination survey and label the bag to indicate the dose rates and contamination levels. The movement of the pole resulted in contaminating a portion of the fuel building. On January 8, 2004, decontamination personnel began to decontaminate the bagged pole in the decontamination booth. When removing the pole from the bag, one worker was internally contaminated, and three other workers were externally contaminated. All of these contamination events were minor and none exceeded regulatory limits. These events were identified by the contamination monitors when the individuals exited the radiologically controlled area.

Analysis. The failure to perform surveys to evaluate the magnitude and extent of radiation levels, and the concentrations or quantities of radioactive materials are performance deficiencies. The finding is greater than minor because it was associated with the Occupational Radiation Safety cornerstone attribute of Program and Process and affected the cornerstone objective to ensure adequate protection of the worker’s health and safety from exposure to radiation. The finding involved individuals’ potential for unplanned or unintended dose and was evaluated with the Occupational Radiation Safety SDP. The finding was determined to be of very low safety significance because it was not associated with an as low as is reasonable achievable planning or work control issue, there was no overexposure or a substantial potential for an overexposure, and the ability to assess dose was not compromised.

Enforcement. 10 CFR 20.1501(a) requires, in part, that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and that are reasonable under the circumstances to evaluate the extent of radiation levels, concentrations or quantities of radioactive materials, and the potential radiological hazards that could be present. Pursuant to 10 CFR 20.1003, a "survey" means an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. 10 CFR 20.1201(a) states, in part, that the licensee shall control the occupational dose to individual adults; however, the licensee failed to perform an adequate evaluation of the potential radiological hazards. Because this failure to perform radiological surveys is of very low safety significance and has been entered into the licensee’s corrective action program as SMF-2004-0069 and SMF-2004-1264, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000445; 446/2004003-02, two examples of a 10 CFR Part 20 violation for failure to perform a radiological survey.

3. Introduction. The inspector reviewed a self-revealing Green NCV of Technical Specification 5.4.1a for the failure of an individual to follow a radiation work permit requirement.

Description. During review of corrective action document SMF-2004-1202, it was determined that scaffold builders working in a high radiation area, under Radiation Work Permit RWP-1215, Task 4, built a scaffold inside the Loop 1 and 4 steam generator cubical up into an area in which radiation protection personnel had not surveyed. A survey was performed of the initial area where the scaffold was to be built, and the workers were briefed on the dose rates (10 to 30 millirem per hour.) However, the workers continued to build the scaffold higher than the initial area where the survey was performed without notifying radiation protection to perform another survey. The workers built the scaffold up into an area with elevated dose rates, and a worker received an electronic dosimeter dose rate alarm. Radiation protection performed a survey of the area and identified general area dose rates of up to 150 millirem per hour.

Analysis. The failure to notify radiation protection before building scaffolding into an unsurveyed area is a performance deficiency. This finding is greater than minor because it is associated with the Occupational Radiation Safety cornerstone attribute of Program and Process and affected the cornerstone objective to ensure adequate protection of the worker's health and safety from exposure to radiation. The finding involved individuals' potential for unplanned or unintended dose and was evaluated using the Occupational Radiation Safety SDP. The finding was determined to be of very low safety significance because it was not associated with an as low as is reasonable achievable planning or work control issue, there was no overexposure or a substantial potential for an overexposure, and the ability to assess dose was not compromised.

Enforcement. Technical Specification 5.4.1 states, in part, written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Appendix A, Revision 2. Regulatory Guide 1.33, Appendix A, Section 7e, requires procedures for access control to radiation areas including the radiation work permit system. Radiation Work Permit RWP 2004-1215, Task 4, Work Instruction No. 3, states, "Notify RP prior to the start of scaffold construction, radiological surveys may need to be performed due to changes in elevation." However, the individual failed to notify radiation protection prior to constructing scaffolding in unsurveyed elevations. Because the failure to follow a radiation work permit requirement is of very low safety significance and has been entered into the licensee's corrective action program as SMF-2004-1202, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000445; 446/2004003-03, Technical Specification 5.4.1 violation for failure to follow a radiation work permit requirement.

4. Introduction. The inspector reviewed two examples of a self-revealing Green NCV of Technical Specification 5.7.1e for the failure of personnel to receive a briefing on radiation dose rates prior to entering a high radiation area.

Description. During review of Corrective Action Document SMF-2004-0471, it was determined that an individual entered a high radiation area that is not routinely surveyed without contacting radiation protection and being briefed on the dose rates in the area. The individual entered the Unit 1, Waste Monitor Tank Room X-185, and received a dose rate alarm. The individual exited the room and noticed the posting of the room as "High Radiation Area, Area not routinely surveyed, Survey required prior to entry, Contact RP." The individual contacted radiation protection, and a survey of the room identified general area dose rates of as high as 220 millirem per hour.

In addition, Corrective Action Document SMF-2004-0620 described that an individual entered into the Unit 1, Room 77B, a posted high radiation area, by walking under a posted rope barricade without contacting radiation protection. The individual was stopped by another worker, and the individual left the area. General area radiation levels in the room were as high as 150 millirem per hour.

Analysis. The failure to receive a briefing on radiation dose rates prior to entering a high radiation area is a performance deficiency. This finding is greater than minor because it is associated with the Occupational Radiation Safety cornerstone attribute of Program and Process and affected the cornerstone objective to ensure adequate protection of the worker's health and safety from exposure to radiation. The finding involved individuals' potential for unplanned or unintended dose and was evaluated with the Occupational Radiation Safety SDP. The finding was determined to be of very low safety significance because it was not associated with an as low as is reasonable achievable planning or work control issue, there was no overexposure or a substantial potential for an overexposure, and the ability to assess dose was not compromised.

Enforcement. Technical Specification 5.7.1e states, in part, that entry into high radiation areas shall be made only after dose rates in the area have been determined and entry personnel are made knowledgeable of them; however, individuals failed to receive a briefing of the dose rates within the high radiation areas prior to entry. Because the failure to follow a Technical Specification requirement is of very low safety significance and has been entered into the licensee's corrective action program as SMF-2004-0471 and SMF-2004-0620, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000445; 446/2004003-04, two examples of a Technical Specification 5.7.1e violation for failure of personnel to receive a radiological briefing prior to entering a high 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:

- Current 3-year rolling average collective exposure
- Site specific ALARA procedures
- Three work activities of highest exposure significance completed during the last outage
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Intended versus actual work activity doses and the reasons for any inconsistencies
- Assumptions and basis for the current annual collective exposure estimate, the methodology for estimating work activity exposures, the intended dose outcome, and the accuracy of dose rate and man-hour estimates
- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered
- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

The inspector completed 8 of the required 15 samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors reviewed a sample of the performance indicator data submitted by the licensee regarding the mitigating systems cornerstone to verify that the licensee's data was reported in accordance with the requirements contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 2. The sample included data taken from control room operator logs, limiting conditions for operation logs, and the SMF database. The inspectors interviewed licensee personnel accountable for collecting and evaluating the performance indicator data and a system engineer. The inspectors also reviewed the mitigating systems performance indicator worksheets for the period from January 2003 through March 2004 for both Units 1 and 2 for the following performance indicators:

- High pressure safety injection system unavailability
- Auxiliary feedwater system unavailability
- Emergency AC power system unavailability
- Residual heat removal system unavailability

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety Cornerstone

a. Scope

The inspector sampled licensee submittals for the performance indicators listed below for the period from June 2003 through March 2004. To verify the accuracy of the performance indicator data reported during that period, performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

- Occupational Exposure Control Effectiveness Performance Indicator

Licensee records reviewed included corrective action documentation that identified occurrences of locked high radiation areas (as defined in Technical Specification 5.7) and very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 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.

b. Findings

No findings of significance were identified

.3 Public Radiation Safety Cornerstone

a. Scope

The inspector sampled licensee submittals for the performance indicators listed below for the period from June 2003 through March 2004. To verify the accuracy of the performance indicator data reported during that period, performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

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

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.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Semi-Annual Review

a. Inspection Scope

On June 23, 2004, the inspectors completed a semi-annual review of licensee internal documents, reports, audits, and performance indicators to identify trends that might indicate the existence of more significant safety issues. The inspectors reviewed the following documents:

- Corrective Action Documents
- System Health Reports
- Self-Assessment and Benchmarking Program Health Report, 1st Quarter 2004
- Corrective Action Program Health Indicators Package, 1st Quarter 2004
- Top Ten Engineering Issues List 2004
- Top Ten Systems With the Lowest Design Margins List
- Nuclear Overview Department (NOD) Outage Review Audit
- NOD Common Cause Analysis Audit Improvement Recommendations

b. Findings and Observations

No findings of significance were identified. However, during the review, the inspectors observed the following issues which were all entered into the corrective action program:

- According to the corrective action program health indicators, there was a decline in the proper prioritization of smart forms for the first quarter 2004. This issue was placed into the corrective action program as SMF-2003-001029-00.

- During a recent performance of an NOD evaluation, EVAL-2004-013, “Safety Tagging Process Review,” it was determined that some maintenance personnel do not understand portions of the clearance/safety tagging program implementation. Also, maintenance personnel did not indicate a low threshold for smart form generation for maintenance tagout issues. These issues were placed into the corrective action program as SMF-2004-002134-00.

- NOD Audit 2004-04, "Common Cause Analysis,” which included team members from other Strategic Teaming and Resource Sharing Alliance (STARS) plants, identified several areas for improvement. Improvement items included: actual root causes were not being identified, the extent of condition review needs to be improved, and the extent of cause is not being performed well for most root cause analyses. These issues were placed in the corrective action program as SMF-2004-000580-00.

- Self-Assessment and Benchmarking Program Health Report for the first quarter 2004 identified that self-assessment performance was declining. This issue was placed into the corrective action program as SMF-2004-000339-00.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected five SMFs for detailed review (SMF-2004-000049-00, SMF-2004-000070-00, SMF-2004-000618-00, SMF-2004-001142-00, SMF-2004-001193-00). The SMFs were associated with a pressure transmitter test card failure, a failure to flush station service water cross connect, operations department 2003 goals for personnel errors not met, an unexpected annunciator alarmed when solid state protection system was deenergized, and a valve that failed causing loss of reactor coolant pump seal injection flow. The reports 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 inspectors evaluated the reports against the requirements of the licensee’s corrective action program and 10 CFR Part 50, Appendix B.

b. Findings and Observations

There were no findings identified associated with the five reviewed samples. The inspectors verified that the root cause evaluation and associated corrective actions were appropriate and also timely, relative to the identified problem; therefore no violation of regulatory requirements or findings were identified.

.3 ALARA Planning and Controls

Section 2OS2 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding access controls to radiologically significant areas exposure tracking, higher than planned exposure levels, and radiation worker practices.

The inspector reviewed the corrective action documents listed in the attachment against the licensee's problem identification and resolution program requirements. No findings of significance were identified.

4OA3 Event Followup (71153)

1. (Closed) Licensee Event Report (LER) 50-446/2003-003-00, Containment Spray Additive System Inoperable Due to Mispositioned Valves

On November 2, 2003, the licensee discovered Unit 2 Train B Valves 2CT-0034 and 2CT-0030 mispositioned. Plant equipment operators left the valves closed on October 21, 2003, after removing a clearance that instructed them to place the two valves in the open position. While in the closed position, these two valves caused one chemical eductor to be inoperable which caused the containment spray additive system to be inoperable. The valves were closed for longer than the allowed outage time in Technical Specification 3.6.7. The licensee attributed the cause to a lack of specific training and procedural guidance for verifying the status of this type of valve which has a remote "knocker" operating mechanism. The licensee took the following corrective actions: immediately opened the two valves to restore the spray additive system to an operable status, issued a Lessons Learned report to shift operations personnel, and performed a position verification lineup on similar remotely operated "knocker" valves in Units 1 and 2 to confirm that the valves were in the correct position. No other mispositioned valves were identified. The licensee performed training on this event and revised the applicable operations procedures to provide operations personnel additional information on the operation of "knocker" type valves. This finding is more than minor because it impacted safety in that it affected the ability to scrub radioactive iodine from containment in the event of a loss of coolant accident. The finding affects the Barrier Integrity cornerstone and has very low safety significance (Green) using the SDP for Reactor Inspection Findings for At-Power Situations because there was not an actual open pathway in the physical integrity of reactor containment nor an actual reduction of the atmospheric pressure control function of the reactor containment. This licensee-identified finding violated Technical Specification 3.6.7, Spray Additive System. The enforcement aspects of this violation are discussed in Section 4OA7. This LER is closed.

.2 (Closed) LER 05000446/2004-001-00: Refueling Water Storage Tank Level Channel Inoperable Due to Isolation of Reference Leg.

On January 13, 2004, while investigating a low level alarm, operators discovered an isolation valve in the reference leg to Unit 2 Refueling Water Storage Tank Level Transmitter 2-LT-0932 closed. The valve had been left closed following a sensor response time test performed on January 6, 2004. With the valve closed, the level channel had been inoperable for longer than allowed by Technical Specifications 3.3.2. The licensee determined the cause of this event was the failure of the technician to restore the valve following the test. The test procedure had not been updated following a modification that added the valve, and the technician had not stopped the test to have the procedure corrected. The corrective actions included procedure revisions and reinforcing management's expectations. No new findings were identified in the inspector's review. This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee documented this event in SMF 2004-000100-00. This LER is closed.

4OA4 Cross Cutting Aspects of Findings

Section 2OS1 described an issue with human performance cross-cutting aspects which involved an entry into a high radiation area without a briefing on radiation dose rates.

4OA5 Other Activities

1. Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02) (Temporary Instruction 2515/152, Revision 1)

This Temporary Instruction provided guidelines to verify compliance with licensee commitments to NRC Bulletin 2003-02, "Leakage From Reactor Pressure Vessel (RPV) Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity." The inspectors used the criteria for bare metal visual examination to conduct this inspection on the CPSES Unit 1 RPV lower head during the 1RF10 refueling outage, Spring 2004.

a. Inspection Scope

The inspectors performed this performance-based evaluation and assessment to ensure that the NRC had an independent review of the condition of the RPV lower head and the Bottom Mounted Instrumentation (BMI) tube penetrations. The inspectors assessed the effectiveness of the licensee examinations of the reactor vessel BMI penetrations. Specifically, the inspectors:

- met with licensee representatives to review inspection plans
- attended pre-job briefs
- directly inspected and assessed the condition of the RPV lower head and the BMI tube penetrations
- reviewed a large representative sample of the visual inspection from inside the reflective metal insulation via a video camera delivered by a remote controlled inspection robot
- assessed the physical difficulties in performing the inspection, which included any debris, dirt, boron, and other viewing impediments
- interviewed the examiner, the equipment operators, and the inspection robot representative
- assessed the licensee's ability to distinguish small boron deposits on the RPV lower head
- evaluated the quality and resolution of the examination equipment

- reviewed completed records, including the final engineering inspection report for CPSES Unit 1
- verified that the licensee documented deficiencies in their corrective action program
- assessed the overall effectiveness of the process used to perform the bare metal visual inspection

The inspectors also reviewed the following documents during this inspection:

- NRC Bulletin 2003-02, "Leakage From Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," dated August 21, 2003
- Comanche Peak Steam Electric Station 30-Day Response to NRC Bulletin 2003-02, "Leakage From Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," TXX-03163, dated September 19, 2003
- Comanche Peak Steam Electric Station 60-Day Response to NRC Bulletin 2003-02, "Leakage From Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," and report on RCS Conoseal Leakage, TXX-03195, dated December 18, 2003
- Comanche Peak Steam Electric Station engineering report, "Unit 2 Baseline Reactor Vessel Lower Bare-Metal Visual Inspection," dated December 15, 2003
- Comanche Peak Steam Electric Station Units 1 & 2 inspection plan, "Reactor Vessel Lower Head Visual Inspection Plan," Revision 0, dated August 28, 2003
- NRC Information Notice 2003-01, "Leakage Found on Bottom-Mounted instrumentation Nozzles," dated August 13, 2003

b. Findings

No findings of significance were identified. The inspectors concluded that the licensee met the applicable commitments in that they performed an inspection of the RPV lower head and 100 percent of the circumference of all 58 BMI tube penetrations and the inspection was performed by a VT-2, Level III certified examiner. The clarity and resolution of the examination equipment, combined with the training, qualification, and procedures, ensured that the examiners could detect small boron deposits. The inspectors have provided the following details of the inspection as required by Temporary instruction 2515/152, "Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC BULLETIN 2003-02)," Revision 1, dated November 5, 2003.

1. Examination

The licensee's examiner was certified in accordance with CPSES procedures to meet the ASME Section XI for VT-2 Level III. The licensee used a tetherless robot to perform the major part of the reactor vessel lower head inspection along with direct inspection as a contingency and to supplement the tetherless robot. The equipment operators exercised the inspection robot on a full-scale mockup built at the South Texas Project for repair activities on their cracked BMI tubes.

The examination was conducted in accordance with the "Reactor Vessel Lower Head Visual Examination Plan," Revision 0, approved on August 28, 2003. This plan was derived, in part, from the previously performed RPV vessel upper head inspection plan for Units 1 and 2.

The inspectors verified that the Reactor Vessel Lower head Visual Examination Plan provided: (1) description of the bare metal visual inspection technique, the administration of this inspection, and the expectation of 100 percent inspection coverage; (2) explicit descriptions of the types of boric acid indications that might be identified; (3) types of indications that shall be investigated further, including boric acid buildup, wastage of carbon steel, and evidence of primary water leakage; (4) criteria for cleaning the lower head and general area; (5) acceptance criteria for the inspection; and (6) sufficient guidance to satisfy licensee commitments for the inspection of the RPV lower head penetrations and general surface of the RPV lower head. The inspectors concluded that the inspection plan, combined with the training, had provided adequate guidance for the licensee examiner to identify, disposition, and resolve deficiencies.

The inspectors determined that the robotic inspection coupled with the direct visual inspections of the VT-2 level III examiner enabled easy identification of boundary leakage as described in NRC Bulletin 2003-02 and any RPV lower head corrosion, if present.

2. Capability to identify and characterize small boric acid deposits

The inspectors determined that the visual inspection methods used by the licensee, as described in the following section (3.), were capable of detecting, identifying, and characterizing small boric acid deposits, if present as described in NRC Bulletin 2003-02. This was determined via direct inspection during the licensee visual inspection of the RPV lower head, and by independent review of the video (DVD and VHS tapes) and photographic medium provided by the licensee.

3. Visual inspection protocol

The bare metal visual inspection was conducted by a tetherless (wireless) robot, supplemented by direct visual inspection. All inspections were performed by a VT-2 Level III certified examiner.

The tetherless robot called "FlangeBot," which was initially purchased to clean the reactor vessel flange, performed the majority of the bare metal visual inspection. The FlangeBot is a wheeled robot that operated on the inside of the lower head reflective metal insulation and gives a view from below the BMI tube penetrations. The video camera aboard the FlangeBot had tilt, zoom and lighting capabilities. The resolution of this camera was verified at six inches and at five feet (expected distances from the insulation to the BMI tube penetrations) with a neutral gray test card, a Jaeger Character Resolution Card, and a color chart. At a distance of six inches the J1 (character height 0.021") characters were readable, and at a distance of five feet, the J1 characters were discernable but the J2 (character height 0.042") characters were readable. The required resolution of a character height of 0.158 inches, per the inspection plan, was demonstrated. The FlangeBot inspection results were documented on a DVD along with verbal annotation.

Direct visual inspection was performed by a VT-2 Level III examiner during equipment installation, during the robotic inspection, and during equipment removal. Direct visual inspection was also used as an initial evaluation of the RPV lower head, and during parts of the robotic evaluation.

4. Inspection coverage

The inspectors determined that the licensee was able to fulfill their commitment to the NRC by completing a 100 percent, 360 degree bare metal visual inspection of the reactor vessel lower head and all 58 BMI tube penetrations.

5. Condition of reactor pressure vessel lower head

In general, the examinations revealed that the RPV lower head was in good physical condition and was observed to be fairly clean, but did exhibit evidence of water flow from sources above the BMI penetrations. In comparison with the conditions observed in the Unit 2 BMI region, Unit 1 appeared to have more evidence of leakage on the vessel itself and not have as much evidence on the reactor vessel insulation. Evidence of this consisted of inactive flow trails that were widely distributed about the reactor vessel circumference and were clearly traceable to previous leaks in the reactor refueling cavity manhole penetrations. The flow trails deposited a thin, two dimensional film of rust-colored material as the trails dried and no appreciable buildup of this material on the lower RPV head or BMI penetration tubing was observed. There were several instances where these trails intersected BMI tubes, collected, and ran down the downhill side of the tubes. Some of these tubes in this population appeared to have their annulus regions bridged by a rust-colored residue. This rust-colored material did not demonstrate the characteristics of an RCS leak and did not appear to emerge from the annulus region. Upon closer review, there was evidence, on the horizontal section of insulation below these tubes, of small pools of borated water that had dried and left white boric acid type of residue. This evidence supports the explanation of the rust-colored material in the annulus as a result of previous leaks from above the lower RPV head.

As noted in the previous inspection of the Unit 2 lower RPV head area, some of the BMI tubes had small white marks located on the tube base and very near the annulus between the reactor vessel and the BMI tube. The marks on these BMI tubes did not have a connection to the annulus region, were two-dimensional, and had a well defined shape with sharp, smooth edges, but within these edges the marks are neither continuous or solid. These features suggest a manual process and do not indicate a deposit emanating from the penetration annulus or from a tight crack through the base metal. Industry experience has demonstrated that these types of RCS leaks have definite three-dimensional characteristics, which would indicate an uncontrolled natural process.

The reactor vessel lower head also seemed to have been painted with a grey coating. Historical photographic data shows that when the reactor vessel arrived at the site it was black in color. The current BMI inspection found that the RPV lower head had a grey coating and the black (brownish-black) coloration was observed to begin just outside the outer BMI tube penetrations and continuing up the reactor vessel. Evidence to the RPV lower head coating was also seen around the base of the BMI tube penetrations. Many of the BMI tube penetrations had an angular hexagonal ring around the penetrations that exposed a brownish-black color, which closely resembled the color of the upper reactor vessel seen outside the outer BMI tubes. This suggested that the tubes were masked in preparation for coating application. A few horizontal grey streaks were discovered on several tubes that resembled a paintbrush stroke where masking was deficient.

The condition of the lower RPV head insulation was not as good as in Unit 2 and had areas that were rough and uneven. There was also quite a bit more debris on the horizontal parts of the insulation. Some of the material looked old, black and was mainly comprised of small particles. This debris was concentrated in the region outside the lower RPV head circumference, but was located in the surrounding regions as well.

6. Identified material deficiencies that required repair

No material deficiencies that required repair were identified.

7. Impediments to effective examinations

The inspectors concluded that, in general, the licensee encountered no serious impediments to performing a 100 percent bare metal examination of the RPV lower head and the BMI tube penetrations. The rough insulation did present some challenges to the FlangeBot, but the skill of the operators compensated for this problem. The licensee’s preparation, training and experience coupled with the available lighting, the excellent quality of the remote controlled robot, equipment, and camera resolution provided a thorough, complete, and well documented inspection.

8. Follow-up examinations above reactor pressure vessel lower head

The licensee did perform appropriate follow-up Alloy 600 weld and pipe inspections in areas above the RPV lower head, which included vessel hot and cold leg nozzle penetration areas. Entrance to these reactor vessel hot and cold leg penetrations was through the manways in the reactor vessel refueling cavity. Removal of these manways revealed evidence of past leakage through these openings. Inspections and photographs of the hot and cold leg penetrations show evidence of past leaks from the reactor cavity manways. Evidence includes small accumulations of boric acid, residual water marks, and staining. The hot and cold leg nozzles were intact and showed no sign of leakage. The inspectors concluded that leakage from the reactor cavity manways was the cause of the flow stains and small accumulations on the lower head and insulation.

9. Samples of deposits and chemical analysis

As described in the previous section (5. and 8.) there were areas on the reactor vessel lower head that exhibited signs of a previous leak from above. There were no indications of deposits or evidence of primary coolant leaks on the reactor vessel lower head or in the annulus regions of the BMI tube penetrations. None of the stains on the reactor vessel lower head amounted to a collectable amount, they were two-dimensional in nature, and were indicative of previous leaks from above the RPV lower head. The licensee acted according to the licensee approved and NRC reviewed reactor vessel lower head inspection plan, and no samples were extracted and, therefore, no chemical analysis was performed.

10. Plans for cleaning of the reactor pressure vessel lower head

The licensee currently has no plans to clean the reactor vessel lower head or the reflective metal insulation. The basis for not cleaning the reactor vessel lower head was that the amount of material on the RPV lower head was small, dry, and very thin (stained), did not impede current or future inspection of this area, and was not perceived as a threat to the carbon steel vessel material. The basis for not cleaning the reflective metal insulation was that the material residue posed no threat to the insulation itself or the reactor vessel, and did not impede inspection of the reactor vessel lower head area. In both cases, the benefit of cleaning the reactor vessel lower head area and the reflective metal insulation would not outweigh the expected dose received by cleaning personnel. The licensee demonstrated the application of good ALARA principles and practices.

11. Licencee's conclusions regarding deposit origins

The licensee has concluded that the origins of the minimal amount of deposit material on the reflective metal insulation and the flow trails on the lower head itself was attributed to previous leaks in the reactor cavity manway seals. The licensee determined this through a series of follow up inspections of the hot and cold leg penetration areas. The licensee determined that the deposits and flow trails were not indicative of RCS leakage or lower head degradation, and were not deemed to be an impediment to current or future inspection activities.

2. <u>Offsite Power System Operational Readiness (Temporary Instruction 2515/156)</u>

This Temporary Instruction requested information to confirm, through inspection and interviews, the operational readiness of offsite power systems in accordance with NRC requirements prescribed in Appendix A to 10 CFR Part 50, General Design Criterion 17 - Electric Power Systems; Appendix B of 10 CFR Part 50, Criterion XVI - Corrective Actions; Technical Specifications 6.8.1 - Electric Power Systems, AC Sources - Operating; 10 CFR 50.63, Loss of all alternating current power; and 10 CFR 50.65(a)(4), assessment and management of risk for maintenance activities. The answers to the questions in the attachment to the Temporary Instruction were transmitted to the Office of Nuclear Reactor Regulation as specified in the Temporary Instruction.

a. Inspection Scope

The inspector interviewed personnel in operations, work control, system engineering, risk and reliability engineering, regulatory affairs, and nuclear overview. The inspector reviewed SMFs related to the dual unit trip in May 2003 and the August 2003 loss-of-grid event; station procedures for configuration risk management, work control, switchyard control, post trip evaluations, EDG operability testing, offsite power operability surveillance, alarm response to low voltages; operating guide and operating procedure for Electric Reliability Council of Texas (ERCOT); and TXU Electric Transmission Planning Procedures. Specifically, the inspector reviewed the licensee's procedures and processes to ensure that risk related to switchyard or grid conditions was assessed and managed during maintenance planning and control. The corrective action documents were reviewed to assess the licensee's actions that resulted from lessons learned from site events and industry events. The ERCOT documents were reviewed for consistency with voltage requirements and communication agreements between the site and the transmission and distribution service provider (TDSP). The staff interviews provided the answers to the Temporary Instruction questions. The documents reviewed are listed in the attachment to this inspection report.

b. Findings and observations

No findings of significance were identified. The responses to the "key" questions to assess the offsite power system's operational readiness for the summer 2004 are shown below:

1. Are there agreements to notify CPSES if the grid is stressed to the point a trip of the plant would result in inadequate voltage to the CPSES switchyards?

   Yes, the ERCOT operating guidance requires the TDSP to monitor real time voltages and provide notice to the qualified scheduling entity (QSE) representing CPSES of any voltage inadequacies at the CPSES switchyards that cannot be corrected within 30 minutes. The QSE notifies the CPSES control room of grid conditions that adversely impact the switchyard. This information is evaluated for risk impact on scheduled and emergent work, such as EDG surveillance testing.

2. Does the agreement between the TDSP and CPSES include the required voltage range and the post-trip load from the plant that will be connected to the grid?

   Yes, the ERCOT operating procedures contain the required CPSES switchyard voltage limits that must be maintained by the transmission system. These limits ensure the safety related bus voltages are maintained within the required range to remain operable. The post trip loads on the startup transformers have also been provided to ERCOT. An annual study is required by ERCOT to evaluate specific contingencies that may impact CPSES switchyard voltage to assure that these voltages remain within these limits. The contingencies include simultaneous loss of a CPSES unit and the most critical transmission line to CPSES.

3. How often is the post trip switchyard voltage calculated?

   ERCOT performs a state estimator analysis of the grid every five minutes to identify transmission line congestion. This analysis includes the possible loss of a critical transmission line or trip of a generating unit. If this calculation indicated a grid condition that would not satisfy the voltage requirements for the CPSES switchyards, the QSE would notify CPSES.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The inspectors presented the results of the inservice inspection effort to Mr. J. Finneran, Acting Technical Support Manager, and other members of licensee management on April 2 and April 9, 2004. Licensee management acknowledged the inspection results. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

On April 9, 2004, the inspector presented the Access Control to Radiologically Significant Areas inspection results to Mr. R. Flores, Vice President, Operations, and members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On May 28, 2004, the inspector presented the ALARA Planning and Controls inspection results to Mr. R. Flores, Vice President, Operations, and members of his staff who acknowledged the findings. On June 8, 2004, the inspector conducted additional discussions by telephone with Messrs. T. Hope, Manager, Regulatory Performance, and D. Wilder, Manager, Radiation Protection & Safety Services. The inspector confirmed that proprietary information was not provided or examined during the inspection.

The inspectors presented the integrated resident inspection results to Mr. J. Kelley, Vice President, Nuclear Engineering and Support, and other members of licensee management on June 24, 2004. The licensee acknowledged the findings presented. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- Technical Specifications 3.6.7 requires that the spray additive system be restored to operable within 72 hours if inoperable. On November 2, 2003, the system had been inoperable for greater than 72 hours. This was documented LER 50-446/2003-003-00 and in the licensee corrective action program as SMF-2003-3559. This finding is of very low safety significance because it did not represent an open pathway in the physical integrity of the reactor containment nor a failure of the pressure control function inside containment. Refer to Section 4OA3.1.

- Technical Specification 5.7.1.d requires, in part, that each individual or group entering high radiation areas (greater than 0.1 rem per hour to 1 rem per hour at 30 centimeters) possess a radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached. On April 18, 2004, an individual entered a high radiation area (with dose rates as high as 150 millirems per hour at 30 centimeters from the source of radiation) in the Unit 1 Pressurizer cubical without logging into the dose tracking computer system. The individual was there for approximately 15 minutes without an operating electronic dosimeter. The occurrence was documented in SMF-2004-1538. This finding is of very low significance 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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

I. Barnes, Steam Generator Consultant
O. Bhatty, Senior Engineer
W. Black, Maintenance Rule Coordinator
M. Blevins, Senior Vice President & Principal Nuclear Officer
M. Bozeman, Manager, Emergency Preparedness
S. Bradley, Supervisor, Health Physics, Radiation Protection & Safety Services
J. Curtis, Manager, Radiation Protection, Radiation Protection & Safety Services
E. Dalasta, Engineer, Repair and Replacement
E. Dyass, Auditor, Quality Assurance
S. Ellis, System Engineering Manager
J. Finneran, Acting Manager, Technical Support
R. Flores, Vice President Operations
C. Harrington, Technical Support Engineering
T. Hope, Manager, Regulatory Performance
J. Kelley, Vice President, Nuclear Engineering and Support
S. Lakdawala, Supervisor, Engineering Programs
M. Lucas, Director of Nuclear Engineering
F. Madden, Regulatory Affairs Manager
T. Marsh, Work Control Manager
D. Mcgaughey, Operations Training Supervisor
R. Morrison, Maintenance Smart Team Manager
P. Passalugo, Inservice Inspection Coordinator
J. Reagan, Level III Nondestructive Technician
J. Skelton, System Engineer
R. Smith, Operations Manager
K. Strider, System Engineer
R. Tapia, System Engineer
D. Wilder, Radiation and Industrial Safety Manager, Radiation and Industrial Safety
C. Wilkerson, Senior Engineer, Regulatory Affairs

Contractors

J. Hair, Authorized Nuclear Inservice Inspector
G. Morini, Project Manager, Wesdyne
V. Polizzi, System Engineer, Westinghouse

NRC Personnel

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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000445, 446/2004003-01	VIO	Entry into a high radiation area without a briefing on radiation dose rates (Section 2OS1)
05000445, 446/2004003-02	NCV	Two examples of a 10 CFR 20.1501 violation for failure to perform a radiological survey (Section 2OS1)
05000445, 446/2004003-03	NCV	Technical Specification 5.4.1 violation for failure to follow a radiation work permit requirement (Section 2OS1)
05000445, 446/2004003-04	NCV	Two examples of a Technical Specification 5.7.1e violation for failure of personnel to receive a radiological briefing prior to entering a high radiation area (Section 2OS1)

Closed

05000446/2003-003-00	LER	Containment Spray Addition System Inoperable Due to Mispositioned Valves (Section 4OA3)
05000446/2004-001-00	LER	Refueling Water Storage Tank Level Channel Inoperable Due to Isolation of Reference Leg (Section 4OA3)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R08: Inservice Inspection Activities (71111.08)

System	Component/Weld Identification	Examination Method
Safety Injection	SI-1-039-001-S22S	Visual (VT-3)
Safety Injection	SI-1-039-001-S22R	Visual (VT-3)
Seal Injection	TBX-1-4111-FLG (flange bolting)	Visual (VT-1)
Safety Injection	SI-1-039-001-S22R	Dye Penetrant
Pressurizer Relief	TBX-1-4502-12	Dye Penetrant
Pressurizer Relief	TBX-1-4502-28	Dye Penetrant

System	Component/Weld Identification	Examination Method
Residual Heat Removal	TBX-2-2500-42	Ultrasonic
Residual Heat Removal	TBX-2-2501-38	Ultrasonic
Residual Heat Removal	TBX-2-2520-08	Ultrasonic
Residual Heat Removal	TBX-2-2520-22	Ultrasonic
Residual Heat Removal	TBX-2-2520-23	Ultrasonic
Residual Heat Removal	TBX-2-2520-33	Ultrasonic

Procedures

Procedure	Title	Revision
TX-ISI-08	VT-1 and VT-3 Visual Examination	5
TX-ISI-10	Qualification of Ultrasonic Manual Equipment	2
TX-ISI-11	Liquid Penetrant Examination	7
TX-ISI-70	Magnetic Particle Examination	6
TX-ISI-101	Preservice and Inservice Examination Documentation	7
TX-ISI-302	Ultrasonic Examination Of Austenitic Piping Welds	1
STA-731	ASME Section XI Repair and Replacement Activities	5
WPT-16534	Steam Generator Degradation Assessment for 1RF10 Outage	3
WPT-16532	Steam Generator Logic Charts for 1RF10 Outage	0
WPT-16508	Steam Generator Tube Stabilization due to AVB Wear During 1RF10 and 1RF11	0
WPT-16516	Stabilization Criteria for SG Tubes	0
	Mechanical Plug In-Service Visual Inspection Guideline	2

	Unit 1 Steam Generator Eddy Current Analysis Guidelines	1

Condition Reports

SMF-2003-002949-00	SMF-2003-003333-00	SMF-2003-000510-00
SMF-2001-000540-00	SMF-2003-003433-00	SMF-2003-001115-00
SMF-2002-003030-00	SMF-2001-000541-00	SMF-2003-002510-00
SMF-2002-003235-00	SMF-2003-003351-00	SMF-2003-002649-00
SMF-2002-003359-00	SMF-2003-003549-00	SMF-2003-002658-00
SMF-2002-003615-00	SMF-2003-003562-00	SMF-2003-003172-00
SMF-2003-001683-00	SMF-2004-001176-00	SMF-2003-003218-00
SMF-2004-001173-00		SMF-2004-001292-00
SMF-2003-002668-00		

Work Order Packages

WO# 4-02-142566-00
WO# 3-02-337065-01
WO# 4-03-149571-00
WO# 4-03-151087-00
WO# 2-03-147427-00
WO# 3-03-328126-01

Section 2OS1: Access Control to Radiologically Significant Areas (71121.01)

Procedures

STA-650	General Health Physics Plan, Revision 5
STA-656	Radiation Work Control, Revision 11
STA-660	Control of High Radiation Areas, Revision 8
RPI-110	Radiation Protection Shift Activities, Revision 8
RPI-528	Multiple Dosimetry Badging, Revision 8
RPI-606	Radiation Work and General Access Permits, Revision 11
RPI-611	Radiological Controls for Diving Operations, Revision 4

Radiation Work Permits

2004-1215	1RF10 Scaffold Erection for Non High Radiation Areas
2004-1400	RP and Decon Evolutions Required to Support Steam Generator Activities
2004-1401	RCP Seal Inspection Removal/Replacement and Associated Activities
2004-1404	In Service Inspections and Support Activities During 1RF10
2004-1600	RP Coverage and Decon Support for Refueling Activities

Corrective Action Documents (Smart Forms)

2002-2957, 2003-1180, 2003-2919, 2003-3051, 2003-3110, 2003-3171, 2003-3396,2003-3449, 2003-3594, 2003-3813, 2004-0270, 2004-1199, 2004-1204, 2004-1228, and 2004-1237, 2004-

1538

Self-Assessments and Evaluations

SA-2003-087	Analysis of Personnel Contaminations during 2RF07
SA-2004-001	Radiation Protection Department Analysis of Smart Forms-4th Quarter 2003
EVAL-2003-024	Radworker Practices and Contamination Control during 2RF07

Miscellaneous

2002 Radioactive Effluent Release Report

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents (Smart Forms)

2002-2957, 2003-3396, 2003-3449, 2003-3594, 2004-1538

Audits and Self-Assessments

CPSES Nuclear Overview Department Evaluation Reports: 2002-044, 2003-022, 2003-024
CPSES Self-Assessment Report 2003-089, “ALARA Planning”

Radiation Work Permits

2002-1215	Scaffolding (1RF09)
2004-1215	Scaffolding (1RF10)
2003-2215	Scaffolding (2RF07)
2002-1400	Primary-side Steam Generator Activities (1RF09)
2004-1400	Primary-side Steam Generator Activities (1RF10)
2003-2400	Primary-side Steam Generator Activities (2RF07)
2002-1600	Refueling (1RF09)
2004-1600	Refueling (1RF10)
2003-2600	Refueling (2RF07)

Procedures

RPI-606	Radiation Work and General Access Permits, Revision No. 11
STA-123	Pre-Job and Post-Job Briefs, Revision No. 0
STA-657	ALARA Planning/Debriefing, Revision No. 8

ALARA Committee Minutes

2004-02 (March 11, 2004)

Section 4OA5: Offsite Power System Operational Readiness (Temporary Instruction 2515/156)

STA-604, "Configuration Risk Management and Work Scheduling," Revision 5

STA-629, "Switchyard Control," Revision 2

ODA-108, "Post RPS/ESF Actuation Evaluation," Revision 9

OPT-214A, "Diesel Generator Operability Test," Revision 18

OPT-215, "Class 1E Electrical Systems Operability," Revision 12

ABN-601, "Response to a 138/345 KV System Malfunction," Revision 9

WCI-203, "Weekly Surveillances / Work Scheduling," Revision 17

Electric Reliability Council of Texas (ERCOT) ERCOT's Operating Guide

TXU Electric Transmission Planning Procedures, dated December 2001

2003 Assessment of Grid Reliability for Comanche Peak

SMF-2003-001365-00, Dual unit trip on May 15, 2003

SMF-2003-003845-01, Evaluate generic lessons from loss-of-grid event, Northeast, August 2003

LIST OF ACRONYMS

1RF10	Unit 1's tenth refueling outage
ABN	abnormal operating procedure
ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
BMI	bottom mounted instrumentation
CFR	*Code of Federal Regulations*
CPSES	Comanche Peak Steam Electric Station
EPRI	Electric Power Research Institute
ERCOT	Electric Reliability Council of Texas
LER	Licensee Event Report
NCV	noncited violation
NEI	Nuclear Energy Institute
NOD	Nuclear Overview Department
NRC	Nuclear Regulatory Commission
OPT	operability test
QSE	qualified scheduling entity
QTE	Quick Technical Evaluation
RCS	reactor coolant system
RPV	reactor pressure vessel
SDP	significance determination process
SMF	smart form
SOP	system operating procedure
SSC	structures, systems, or components
STARS	Strategic Teaming and Resource Sharing Alliance
TDSP	transmission and distribution service provider
WO	work order