UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-8064

April 30, 2002

C. L. Terry, Senior Vice President
& Principal Nuclear Officer
TXU Generation Management Company LCC,
Managing General Partner for TXU Generation
Company LP
ATTN: Regulatory Affairs Department
P.O. Box 1002
Glen Rose, Texas 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION - NRC INSPECTION REPORT 50-445/01-06; 50-446/01-06

Dear Mr. Terry:

On April 6, 2002, the NRC completed an inspection at your Comanche Peak Steam Electric Station, Units 1 and 2, facility. The enclosed report documents the inspection findings which were discussed on April 11, 2002, with Mr. J. Kelley and 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 this inspection, the inspectors identified two issues of very low safety significance (Green). One of these issues was determined to involve a violation of NRC requirements. A third issue was identified and determined to be a No Color violation. Because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these violations as noncited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these noncited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response 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

Enclosure:
NRC Inspection Report
50-445/01-06; 50-446/01-06

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-445, 50-446
Licenses:	NPF-87, NPF-89
Report:	50-445,446/01-06
Licensee:	TXU Generation Company LP
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56, Glen Rose, Texas
Dates:	December 30, 2001 - April 6, 2002
Inspectors:	D. B. Allen, Senior Resident Inspector J. S. Dodson, Project Engineer, Project Branch E P. J. Elkmann, Emergency Preparedness Inspector J. M. Keeton, Project Engineer, Project Branch A R. E. Lantz, Senior Emergency Preparedness Inspector S. C. Schwind, Resident Inspector
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 50-445/01-06; 50-446/01-06

IR 05000445-01-06; IR 05000446-01-06; on 12/30/01-04/06/2002;TXU Energy; Comanche Peak Steam Electric Station; Units 1 & 2. Integrated Resident & Regional Report: Operability Evaluations, Postmaintenance Testing, ALARA Planning & Control.

The inspection was conducted by resident inspectors, regional emergency preparedness inspectors and a regional project engineer. The inspection identified two Green findings, one of which was a noncited violation. A third issue was identified and determined to be a No Color noncited violation. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. A noncited violation of TS 5.4.1a. was identified for an inadequate procedure for performing an equalizing charge on safety-related batteries. On January 9, 2002, electrolyte overflowed from two cells of the Unit 1 Train A 125 Vdc Battery BT1ED1 during an equalizing charge, even though the procedure in use contained precautions to prevent the overflow. The procedure did not contain sufficient guidance to ensure the electrolyte levels were monitored frequently enough to avoid overflowing the battery case.

  This violation was more than minor because the overflow condition had an actual impact on safety in that it caused the battery to be inoperable in accordance with Technical Specification 3.8.6. Since the finding affected operability, it was analyzed by the Significance Determination Process (SDP). Phase 1 of the SDP screened the safety significance to be very low (Green) because the battery, a mitigating subsystem, was inoperable for only a few minutes which is less than the allowed outage time of 2 hours and there was no actual loss of safety function. Because the finding was of very low safety significance, and the finding was documented in the licensee's corrective action system, this finding is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Smart Form SMF-2002-000084-00 (Section 1R15).

- Green. A Green finding was identified for the failure of the licensee to specify an adequate postmaintenance test following replacement of the current-to-pneumatic (I/P) converter for the Unit 2, Steam Generator 4 atmospheric relief Valve 2-PV-2328. The tests specified by the work order would not have validated the design basis of the supporting air accumulator. Furthermore, the leak rate from the I/P converter had not been identified as a quality attribute when procuring and conducting receipt inspections of new I/P converters.

After the inspector discussed this with the licensee, a pressure drop test was performed prior to declaring the atmospheric relief valve operable, therefore no violation of NRC requirements occurred. However, this finding was greater than minor because it had a credible impact on safety since the maintenance could have rendered the valve incapable of completing its design function during a loss of offsite power. Since this finding did not involve a design or qualification deficiency and did not represent an actual loss of safety function, Phase 1 of the Significance Determination Process characterized the finding to be of very low safety significance. This issue was documented in the corrective action program as Smart Form SMF-2002-000120-00 (Section 1R19).

Cornerstone: Occupational Radiation Safety

- No Color. On March 13, 2002, an NRC inspector observed an individual leaving the protected area and exiting the portal radiation monitor (PM-7) while the monitor was in alarm. The individual did not stop, and when the inspector called the individual to recount he did not return. The individual was stopped by another site employee and returned for a recount. The recount did not detect any radioactive material. Station Administration Procedure STA-654, "Personnel and Discrete Radioactive Particle Contamination Control," Revision 3, requires that if a portal monitor alarm occurs, the individual is to step out and repeat the count. The failure to follow procedural requirements involving a personnel contamination monitor alarm was a violation of Technical Specification 5.4.1a. This is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Smart Form SMF-2002-000777.

  The safety significance of this violation was determined to be more than minor, because not responding to a personnel contamination monitor alarm had a credible impact on a worker's radiation safety. This violation did not affect the cornerstone since there was no impact on radiation monitors (instrumentation and/or personnel dosimetry) related to measuring workers' dose (Section 2OS2).

Report Details

Summary of Plant Status

Unit 1 operated at essentially 100 percent power for the duration of the report period.

Unit 2 began the report period at 100 percent power. The sixth Unit 2 refueling outage (2RF06) began on March 30, 2002, at 11:47 a.m. when the main generator output breakers were opened. The unit remained in the refueling outage at the end of the report period.

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

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdown

a. Inspection Scope

The inspectors conducted partial walkdowns of the following 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 component materiel condition. 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 Trains A and B motor driven auxiliary feedwater pumps on January 17, 2002, while the turbine driven auxiliary feedwater (TDAFW) pump was removed from service for planned maintenance
- Unit 2 Train B residual heat removal system on March 7, 2002
- Unit 2 Train A containment spray system on March 19, 2002

b. Findings

No findings of significance were identified.

.2 Detailed Semiannual Walkdown

a. Inspection Scope

The inspectors conducted a detailed semiannual walkdown of the Unit 1 auxiliary feedwater system. The walkdown was completed on January 8, 2002, using the system operating procedure and system drawings to verify that components were in the correct position, electrical power was available as required, major components were correctly labeled, hangers and supports were functional, support systems were operational, and transient equipment and debris would not interfere with system operation. The materiel

condition of the system components was assessed. In addition, a search of corrective action documents and outstanding work orders was performed to determine if these conditions would have had an adverse impact on operability of the system. The following documents were used during this inspection:

- System Operating Procedure SOP-304A, "Auxiliary Feedwater System," Revision 14
- Drawing M1-0206, Revision CP-18
- Drawing M1-0206, Sheet 1, Revision CP-12
- Drawing M1-0206, Sheet 2, Revision CP-15
- Auxiliary Feedwater System Health Report for Third Quarter, 2001

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Fire Area Walkdowns

a. Inspection Scope

The inspectors toured the following areas to assess 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. 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 Area 2SK17A,B,C - Unit 2 main steam and feedwater penetration area, on January 10, 2002
- Fire Area AA21B - Auxiliary building 810-foot elevation corridor, on January 10, 2002
- Fire Area AA21D - Auxiliary building 832-foot elevation corridor, on January 23, 2002
- Fire Area 2SB004 - Unit 2 safeguards building 790-foot elevation corridor, on February 21, 2002

- Fire Area 2SG - Unit 2 Train A emergency diesel generator rooms, on January 28, 2002
- Fire Area 2SI - Unit 2 Train B emergency diesel generator rooms, on January 28, 2002

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill

a. Inspection Scope

The inspectors observed the plant fire brigade during a fire drill on February 28, 2002, to assess its ability to fight fires. The fire was simulated in the maintenance annex building where the ambulance and fire truck were parked. Observations focused on the following aspects of the drill:

- Protective clothing/turnout gear was properly donned
- Self-contained breather apparatus equipment was properly worn and used
- Fire hose lines were capable of reaching all necessary fire hazard locations, that the lines were laid out without flow constrictions, and that the hose was simulated being charged with water
- The fire area of concern was entered in a controlled manner (e.g., fire brigade members stayed low to the floor and felt the door for heat prior to entry into the fire area of concern)
- Sufficient fire fighting equipment was brought to the scene by the fire brigade to properly perform their firefighting duties
- The fire brigade leader's fire fighting directions were thorough, clear, and effective
- Radio communications with the plant operators and between fire brigade members were efficient and effective
- Members of the fire brigade checked for fire victims and propagation into other plant areas
- Effective smoke removal operations were simulated
- The fire fighting preplan strategies were utilized

- The licensee planned drill scenario was followed, and that the drill objectives acceptance criteria were met

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 at Comanche Peak. This included a review of flood analysis documentation and calculations to determine areas in the plant susceptible to flooding from internal sources. Based on that review and a review of the Comanche Peak Steam Electric Station (CPSES) probabilistic risk assessment summary document, a walkdown was performed on January 19, 2002, which included the Unit 1 and Unit 2 safeguards building rooms 108A through 108H on the 873-foot elevation, to assess the adequacy of flood protection measures regarding a postulated main feed line rupture. The walkdown included determining whether mitigating systems defined in the flood analysis were in place and functional.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors independently verified that the licensee properly implemented 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” for the following equipment performance problems:

- WO 4-01-138995-00, Packing leak on flow control Valve 1-FCV-121
- SMF 2001-2256, Failed diaphragm on Valve 2-HV-2397-AO
- SMF 2001-2361, Failed power supply in power range Instrument N-44
- SMF 2001-2358, Unit 2, Train B safety chiller trip
- SMF 2001-2722, Foreign material found in instrument air Compressor 2-01 causing low oil pressure
- SMF 2001-2865, Instrument air Compressor 2-02 tripped on motor overload following annual compressor maintenance

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 SSC (if applicable).

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- Planned maintenance on the Unit 1 TDAFW pump concurrent with testing of the anticipated transient without scram mitigation system actuation circuitry on January 10, 2002
- Corrective maintenance to replace pressure regulator on motor driven auxiliary feedwater Pump 2-02 to condensate storage tank recirculation flow Valve 2-FV-2457 on February 21, 2002
- Evaluation 2002-370-03 to assess the impact on Unit 1 of the Unit 2 outage of both trains of component cooling water and both emergency diesel generators during the core offload period of 2RF06
- 2RFO6 steam generator nozzle dam installation including midloop operation with fuel in the vessel
- Outage of the west bus of the 345 KV switchyard during 2RF06 with Unit 1 in Mode 1 on April 5, 2002
- Initial outage risk assessment and management assessment of 2RF06 and subsequent updates through April 5, 2002

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed selected operability evaluations conducted by the licensee during the report period 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 the licensee considered all other pre-existing conditions, as applicable. Additionally, the inspectors evaluated the adequacy of the licensee's problem identification and resolution program as it applied to operability evaluations. Specific operability evaluations reviewed are listed below.

- Design Change Notice 13173, Revision 0, Additional guidance in performing single cell battery charges
- Quick Technical Evaluation 2002-000417-01-00, Discrepancies identified in the review of Westinghouse Technical Bulletin ESBU-TB-96-03-R0
- Smart Form SMF-2002-000561-00, Atmospheric relief valve (ARV) air accumulator check valve test acceptance criteria
- Smart Form SMF-2002-000489-00, Steam generator narrow range level indication at Diablo Canyon did not respond as expected when a feedwater regulating valve failed closed (Operating Experience OE 13294 and Nuclear Safety Advisory Letter 02-003)
- Smart Form SMF-2002-000560-00, During OPT-206B, "TDAFW Pump 2-01," Step 8.3.3 was not satisfied. The step required the turbine to accelerate to greater than or equal to 4075 rpm as indicated on Indicator 2-SI-2452A, AFWPT SPD. The turbine accelerated to 4054 rpm

b. Findings

On January 9, 2002, electrolyte overflowed from two cells of the Unit 1 Train A 125 Vdc Battery BT1ED1 during an equalizing charge. The failure to adequately perform the equalizing charge procedure was determined to be a Green noncited violation.

Procedure MSE-C0-5903, "Battery Maintenance - Equalizing Charge," Revision 4, and Design Change Notice 13173 contained precautions, limitations, and caution statements to interrupt charging if electrolyte overflow was in danger of occurring, although no guidance was provided in the procedure for the frequency of monitoring electrolyte levels. After the first four hourly sets of pilot cell data were recorded, the next three required sets of pilot cell data were not recorded. Battery cells 20 and 30 were subsequently found overflowing approximately 13 hours after the equalizer charge began.

This finding is more than minor because the overflow condition had an actual impact on safety in that it caused the battery to be inoperable in accordance with Technical

Specification 3.8.6. Since the finding affected operability, it was analyzed by the SDP. Phase 1 of the SDP screened the safety significance to be very low (Green) because the battery, a mitigating subsystem, was inoperable for only a few minutes which is less than the allowed outage time of 2 hours and there was no actual loss of safety function.

The failure to limit the charging rate to avoid the overflow of the electrolyte was due to an inadequate procedure for performing an equalizing change. This was a violation of Technical Specification 5.4.1a. which required written procedures be established, implemented, and maintained covering applicable activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A included operating procedures and maintenance procedures for safety related dc electrical systems. Because the finding was of very low safety significance, and the finding was documented in the licensee's corrective action system, this finding is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Smart Form SMF-2002-000084-00 (NCV 50-445/0106-01).

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

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

- Planned maintenance on the Unit 2, Train A centrifugal charging pump speed changer on January 8, 2002
- Replacement of the I/P converter for the Unit 2, Steam Generator 4 ARV 2-PV-2328 on January 14, 2002
- Replacement of the air booster relay for the Unit 2, Steam Generator 4 ARV 2-PV-2328-BR1 on January 17, 2002
- Elastomer replacement on Valve 2-HV-2460-IPI on February 21, 2002.
- Adjusting set pressure of Unit 2 main steam safety Valves 2-MS-0062 and 2-MS-0133 on March 25, 2002.

In each case, the associated work orders and test procedures were reviewed against the attributes in Inspection Procedure 71111, Attachment 19, to determine the scope of the maintenance activity and determine if the testing was adequate to verify equipment operability.

b. Findings

The inspector identified a Green finding when the licensee failed to specify an adequate postmaintenance test following replacement of the I/P converter for the Unit 2, Steam Generator 4 ARV 2-PV-2328.

On January 14, 2002, the licensee was in the process of replacing the I/P converter for ARV 2-PV-2328 as preventive maintenance. Prior to completion of the work, the inspector reviewed the standard work steps in Work Order 3-00-337832-01 and noted that only two postmaintenance tests were specified; a leakage test using a soapy solution of all the mechanical joints in the air system affected by the maintenance, and a timed valve stroke of ARV 2-PV-2328.

The steam generator ARVs are safety-related, air-operated valves which are supplied by instrument air. Since the instrument air system at Comanche Peak is not safety related, each ARV is provided with a safety-related air accumulator which is designed to supply enough air for 15 valve operations, plus a designed leak rate from the I/P converter, pressure regulator, and the valve's air actuator in the event of a loss of instrument air. A leakage test using a soapy solution of the mechanical joints on the new I/P converter and a single valve stroke with instrument air aligned to the accumulator would not have validated the design basis of the ARV's air system following installation of the new component. Furthermore, the leak rate from the I/P converter had not been identified as a quality attribute when procuring and conducting receipt inspections of new I/P converters. Therefore, a pressure drop test on the accumulator should have been specified by the work order in order to assure that the system's design basis remained valid following the maintenance.

The inspector discussed this with the licensee prior to completion of the maintenance. They agreed with this observation and performed a pressure drop test prior to declaring the ARV operable, therefore no violation of NRC requirements occurred. However, this issue had a credible impact on safety since the maintenance on the ARV air system could have rendered the valve incapable of completing its design function during a loss of offsite power. The ARVs are part of the main steam system which, in this context, is considered a mitigating system since they would be used during a loss of offsite power to perform a plant cooldown. This finding did not involve a design or qualification deficiency and did not represent an actual loss of safety function for the system or of a single train for greater than the technical specification allowed outage time. Since this finding did not screen as potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event, Phase 1 of the SDP characterized the finding to be of very low safety significance (Green). This issue was documented in the corrective action program as Smart Form SMF-2002-000120-00.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated licensee Unit 2 outage activities to ensure that risk was considered in developing the outage schedule, the plant configuration was controlled in

consideration of facility risk, mitigation strategies were developed and 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:

- Defense-in-depth and mitigation strategy review
- Refueling outage schedule and risk assessment review
- Unit shutdown and cooldown
- Decay heat removal
- Electrical power sources
- Midloop 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; the adequacy of acceptance criteria including test frequency and test equipment accuracy, range and calibration; procedure adherence; record keeping; the restoration of standby equipment; test failure evaluations; jumper control (if applicable); and the effectiveness of the licensee's problem identification and correction program. The following surveillance test activities were observed by the inspectors:

- Unit 2, Train A diesel generator monthly surveillance (OPT-214B, Revision 10) concurrently with Train A safeguards slave Relay K609 actuation test (OPT-467B, Revision 3) on January 16, 2002

- Unit 2, Train A diesel generator monthly surveillance (OPT-214B, Revision 10) concurrently with Train A safeguards slave Relay K603 actuation test (OPT-465B, Revision 5) on February 13, 2002

- Unit 2 TDAFW Pump 2-01 surveillance (OPT-206B, Revision 12) on February 21, 2002

- Unit 1 TDAFW Pump 1-01 surveillance (OPT-206A, Revision 20) on February 28, 2002

- Unit 1, Train A diesel generator surveillance test (OPT-214A, Revision 14) on March 6, 2002

- Unit 2, Train A residual heat removal pump operability test (OPT-203B, Revision 10) on March 7, 2002
- Unit 2, Train B containment sump inspection (OPT-306, Revision 6) on April 1, 2002
- Unit 2, Train B 125 Vdc Battery BT2ED4 18-month modified performance discharge test (MSE-S0-5715, Revision 1) on April 1, 2002

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

On January 5, 2002, the licensee implemented a temporary modification to one of the Unit 2 containment air cooler condensate flow rate monitors which was believed to be contributing to a nuisance alarm in the control room. This modification was chosen since the containment air cooler condensate flow rate monitor is one of three reactor coolant system leak detection instruments required to be operable per Technical Specification 3.4.15.

The following documents were referenced during this inspection:

- Operations department troubleshooting plan for the Unit 2 vents and drains system dated January 5, 2002
- Operation Department Work Instruction OWI-0007, “Operations Procedure Development and Process Control Review Summary,” Revision 0
- Drawing M2-0238, Sheet A, Revision CP-10
- Smart Form 2002-000029-00

b. Findings

No findings of significance were identified.

1EP2 Alert Notification System Testing (71114.02)

a. Inspection Scope

The inspectors interviewed one licensee emergency preparedness staff member and reviewed Procedure SG-12, “Alert and Notification System Surveillance,” Revision 5, to

determine the adequacy of licensee methods for testing the alert and notification system in accordance with 10 CFR Part 50, Appendix E. The licensee's siren testing program was also compared with NUREG-0654 and Federal Emergency Management Agency REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants."

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspectors interviewed one licensee emergency preparedness staff member and reviewed the following documents related to the emergency response organization augmentation system to determine the licensee's ability to staff emergency response facilities in accordance with the licensee emergency plan and the requirements of 10 CFR Part 50, Appendix E:

- EPP-201, "Assessment of Emergency Action Levels, Emergency Classification, and Plan Activation," Revision 11
- EPP-202,"Emergency Communication Systems and Equipment," Revision 6
- EPP-203, "Notifications," Revision 13
- Temporary Security Instructions, "Site Access Control," Revision 15
- "Site Access During an Emergency," employee email, October 10, 2001
- Emergency Preparedness Bulletin 2001-031

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the following documents related to the licensee's corrective action program to determine the licensee's ability to identify and correct problems in accordance with the requirements of 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E.

- STA-421, "Initiation of Smart Forms," Revision 10

- STA-422, “Processing Smart Forms,” Revision 18
- Evaluation reports for drills and exercises conducted on August 30, 2000, July 18, 2001, August 1, 2001, August 22, 2001, October 24, 2001, and November 13, 2001
- Summaries of 69 corrective actions assigned to the emergency preparedness department between August 2000 and February 2002
- Details of 13 selected Smart Forms
- Self Assessment SA-2001-01, “Verification of NRC Performance Indicators”
- Self Assessment SA-2001-038, “Review of Emergency Planning Surveillance Methodology for the Alert and Notification System Reliability”
- Quality Assurance Overview Strategy Planning Sheets for the first and second quarters of calendar year 2001
- CPSES Nuclear Overview Department Evaluation Reports 2001-042 and 2001-049

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs in the radiologically controlled areas during normal operations. Field observations of selected work areas within the radiologically controlled areas were conducted. The following items were reviewed and compared with regulatory requirements to determine whether the licensee had an adequate program to maintain occupational exposure as low as is reasonably achievable (ALARA):

- ALARA program procedures
- Processes used to estimate and track exposures

- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information
- Nine radiation work permit packages (2002-0022, 0024, 0100, 0300, 0301, 2215, 2301, 2400, and 2600) for work activities with the highest personnel collective exposures during the inspection period
- One job (Radiation Work Permit 2002-0300, “Design Mod on Transfer Cart Fuel Bldg. Side”) was observed, and tours were conducted in various areas of the plant
- Use of engineering controls to achieve dose reductions including design change packages (FDA-1999-000241 and FDA-2001-000861) and five temporary shielding request packages (01-25, 02-16, 02-17, 02-19 and 02-23)
- Exposures of selected work groups (radiation protection, decontamination and mechanical maintenance)
- Hot spot tracking and reduction program
- Plant-related source term data, including source term control strategy
- Radiological work planning
- One quality audit (EVAL-2001-039), two ALARA program status reports (third quarter 2001 and fourth quarter 2001) and four self-assessments (SA-2001-050, 063, 064, and 072)
- ALARA committee meeting minutes (2001-06, dated 10/30/01)
- Selected corrective action documents involving the ALARA program and radiation worker practice deficiencies (Smart Forms: SMF-2001-000619, 002174, 002250, 002275, 002308, 002568, 002680, 003005, 2002-000013 and 000116)
- Declared pregnant worker dose monitoring controls

b. Findings

A noncited violation with No Color was identified for failure to acknowledge a personnel contamination monitor alarm. On March 13, 2002, an NRC inspector observed an individual leaving the protected area and exiting the portal radiation monitor (PM-7) while the monitor was in alarm. The individual did not stop, and, when the inspector called the individual to recount, he did not return. The individual was stopped by another site employee and returned for a recount. The recount did not detect any radioactive material. Station Administration Procedure STA-654,“Personnel and Discrete Radioactive Particle Contamination Control,” Revision 3, requires that if a portal monitor alarm occurs, the individual is to step out and repeat the count. The failure to follow procedural

requirements involving a personnel contamination monitor alarm was a violation of Technical Specification 5.4.1a.

The safety significance of this violation was determined to be more than minor, because not responding to a personnel contamination monitor alarm had a credible impact on a worker's radiation safety. This violation did not affect the cornerstone since there was no impact on radiation monitors (instrumentation and/or personnel dosimetry) related to measuring a worker's dose.

Technical Specification 5.4.1a. requires that written procedures shall be established, implemented, and maintained for the procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Section 7.e(7), requires procedures for personnel monitoring. Station Administration Procedure STA-654, "Personnel and Discrete Radioactive Particle Contamination Control," Revision 3, requires that if a portal monitor alarm occurs, the individual is to step out and repeat the count. The failure to follow procedural requirements involving a personnel contamination monitor alarm was a violation of Technical Specification 5.4.1a. This is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Smart Form SMF-2002-000777 (NCV 50-445; 50-446/0106-02).

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Initiating Events

a. Inspection Scope

The inspector reviewed a sample of performance indicator data submitted by the licensee regarding the initiating events cornerstone to verify that the licensee's data was reported in accordance with the requirements of NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 1. The sample included data taken in 2001 for both units for the following performance indicators:

- Unplanned scrams
- Scrams with loss of normal heat removal
- Unplanned power changes

b. Findings

No findings of significance were identified.

.2 Emergency Preparedness - Drill and Exercise Performance

a. Inspection Scope

The inspectors reviewed the following documents related to the drill and exercise performance indicator in order to verify that the licensee's data was reported in accordance with the requirements of NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 1:

- Drill schedules for calendar years 2001 and 2002
- Drill records for 100 percent of the drills and exercises counted in drill/exercise performance statistics for the third and fourth quarters of calendar year 2001
- Drill evaluation worksheets
- Drill evaluation records
- Performance indicator summary sheets
- Performance indicator reports

b. Findings

No findings of significance were identified.

.3 Emergency Preparedness - Emergency Response Organization Drill Participation

a. Inspection Scope

The inspectors reviewed the following records related to emergency response organization participation in order to verify the licensee's data was reported in accordance with the requirements of NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 1:

- Emergency response organization rosters for the third and fourth quarters of calendar year 2001
- List of key emergency response organization positions
- Drill participation records for a sample of 12 emergency response organization members
- Performance indicator summary sheets
- Performance indicator reports

b. Findings

No findings of significance were identified.

.4 Emergency Preparedness - Alert and Notification System

a. Inspection Scope

The inspectors reviewed siren testing records for the third and fourth quarters of calendar year 2001 to verify the licensee's data was reported in accordance with the requirements of NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 1.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

The inspectors presented the emergency preparedness program baseline inspection results to Mr. M. Blevins, Deputy to Senior Vice President & Principal Nuclear Officer, and other members of licensee management at the conclusion of the inspection on February 14, 2002. The licensee acknowledged the findings presented.

The inspector presented the ALARA planning and controls inspection results to Mr. S. Ellis, Operations Manager, and other members of licensee management at an exit meeting on March 14, 2002. The licensee's management acknowledged the findings presented.

The inspector discussed, by telephone, with Mr. R. Walker, Regulatory Affairs Manager, and other licensee personnel on March 22, 2002, that the failure to acknowledge a personnel contamination monitor alarm had been classified as a No Color violation.

The inspectors presented the resident inspection results to Mr. J. Kelley, Vice President - Nuclear Engineering and Support, and other members of licensee management at the conclusion of the inspection on April 11, 2002. The licensee acknowledged the findings presented.

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

ATTACHMENT

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Alldredge, Supervisor, Radiation Protection - ALARA
M. Blevins, Deputy to Senior Vice President & Principal Nuclear Officer
D. Bozeman, Manager, Emergency Planning
S. Bradley, Supervisor, Health Physics
J. Curtis, Manager, Radiation Protection
S. Ellis, Manager, Operations
B. Hise, Supervisor, Safety Services
J. Kelley, Vice President, Nuclear Engineering and Support
R. Kidwell, Licensing Engineer
D. Moore, Plant Manager
D. O'Connor, Supervisor, Radiation Protection
R. Walker, Manager, Regulatory Affairs
D. Wilder, Manager, Radiation and Industrial Safety
C. Wilkerson, Senior Licensing Engineer

NRC

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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed During this Inspection

50-445/0106-01	NCV	Inadequate procedure for equalizing charge resulted in overflowing two cells of safety-related 125 Vdc battery (Section 1R15).
50-445/0106-02	NCV	Failure to acknowledge a personnel contamination monitor alarm (Section 2OS2).
50-446/0106-02	NCV	Failure to acknowledge a personnel contamination monitor alarm (Section 2OS2).