UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

July 27, 2005

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000445/2005003 AND 05000446/2005003

Dear Mr. Blevins:

On June 23, 2005, 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 29, 2005, with you 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 this inspection, the NRC has identified two issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these issues. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. These NCVs are described in this inspection report. If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region IV; and the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at the Comanche Peak Steam Electric Station facility.

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/

Thomas R. Farnholtz, Chief
Project Branch A
Division of Reactor Projects

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

Enclosure:
NRC Inspection Report 05000445/2005003
and 05000446/2005003
w/attachment: Supplemental Information

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SISP Review Completed: __TRF__ ADAMS: / Yes ☐ No Initials: _TRF____
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RIV:RI:DRP/A	SRI:DRP/A	PE:DRP/A	C:DRS/EB	C:DRS/OB	C:DRS/PEB
AASanchez	DBAllen	MABrown	JAClark	ATGody	LJSmith
E-TRFarnholtz	***E-TRFarnholtz***	***/RA/***	***DLProulx***	***RELantz***	***/RA/***
7/27/05	7/27/05	7/27/05	7/26/05	7/25/05	7/22/05

C:DRS/PSB	C:DRP/A				
MPShannon	TRFarnholtz				
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7/26/05	7/27/05				

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-445, 50-446
Licenses:	NPF-87, NPF-89
Report:	05000445/2005003 and 05000446/2005003
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, 2005
Inspectors:	D. Allen, Senior Resident Inspector A. Sanchez, Resident Inspector T. Farnholtz, Senior Project Engineer T. Brown, Project Engineer C. Paulk, Senior Reactor Inspector, Engineering Branch L. Carson, Senior Health Physicist, Plant Support Branch D. Carter, Health Physics Inspector, Plant Support Branch T. McKernon, Senior Operations Engineer J. Keeton, Consultant
Approved by:	T. R. Farnholtz, Chief, Project Branch A Division of Reactor Projects
Attachment:	Supplemental Information

SUMMARY OF FINDINGS

IR 05000445/2005003, 05000446/2005003; 03/25/2005-06/23/2005; Comanche Peak Steam Electric Station, Units 1 and 2; Licensed Operator Requalification and Event Follow-Up.

This report covered a 3-month period of inspection by two resident inspectors, one regional senior reactor inspector, two health physics inspectors, one senior operations engineer, two regional project engineers, and a consultant. Two 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: Mitigating Systems

- Green. A self-revealing noncited violation was identified for the failure to protect the integrity of the annual reactor operator requalification examination as described in 10 CFR 55.49. The examination material was inadvertently left in the control room simulator facility following annual requalification examination administration. The material was subsequently discovered by the oncoming initial operator licensing instructors. The licensee has counseled individuals involved, reviewed and made changes to the controlling procedure, and reviewed the operator examination security processes and procedures to identify areas for improvement.

  This finding was determined to be more than minor because, if left uncorrected, the finding could become a more significant safety concern. Based on the results of a significance determination process using Manual Chapter 0609, Appendix I, this finding was determined to have very low safety significance, since compensatory actions were immediately taken upon discovery of the examination compromise. The cause of the finding is related to the crosscutting element of human performance (Section 1R11.3).

- Green. A self-revealing noncited violation was identified for the failure of a plant operator to follow established procedures as required by Technical Specification 5.4.1 when the wrong Unit 2 battery charger was de-energized while the plant was shut down in Mode 6 with the refueling cavity at greater than 23 feet and no core alterations in progress. The battery charger that was inadvertently de-energized was the sole power source of the Unit 2 control room annunciators. All Unit 2 control room annunciators were lost for a period of approximately 7 minutes. Refueling cavity level indication and the plant computer remained operational during the event.

This finding is greater than minor because it affected the configuration control, equipment performance, and human performance attributes of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using the Inspection Manual Chapter 0609 Appendix G shutdown operations significance determination process, the inspectors determined the finding was of very low safety significance because the finding did not: (1) increase the likelihood of a loss of RCS inventory, (2) degrade the licensee's ability to terminate a leak path or add RCS inventory when needed, or (3) degrade the licensee's ability to recover decay heat removal once it is lost. The cause of the finding is related to the crosscutting element of human performance (Section 4OA3.1).

B. Licensee Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

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

CPSES Unit 2 began the period at 95 percent power in a coastdown to Refueling Outage 2RF08. On March 26, Refueling Outage 2FR08 began at 11:46 a.m. when the reactor was manually tripped and the main generator output breakers were opened. The refueling outage ended on April 28, 2005, at 12:59 a.m. when the output breakers were closed. The unit achieved approximately 100 percent power on May 4, 2005, and operated at essentially full power for the remainder of the report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed Abnormal Conditions Procedure (ABN) ABN-907, "Acts of Nature," Revision 9, Section 5, "Severe Weather," upon entering a severe thunderstorm and tornado watch on April 25 in response to a National Weather Service advisory. The inspectors reviewed the control room log for activities associated with the ABN-907 entry. The inspectors performed a partial walkdown of the following areas to verify measures in ABN-907 had been implemented prior to the expected onset of severe weather conditions.

- Units 1 and 2 turbine buildings and adjacent exterior and the yard laydown areas used for scaffolding temporary storage

The inspector completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

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

- April 21, 2005, Unit 1 Train A residual heat removal system in accordance with System Operating Procedure (SOP) SOP-102A, “Residual Heat Removal System,” Revision 14, while the Train B residual heat removal system was inoperable for scheduled surveillance

- April 27, 2005, Unit 1 Train B diesel generator system in accordance with SOP-609A, “Diesel Generator System,” Revision 15, while the Train A diesel generator system was inoperable for scheduled maintenance

- June 3, 2005, Unit 2 turbine-driven auxiliary feedwater pump in accordance with SOP-304B, “Auxiliary Feedwater System,” Revision 10, following emergent work to repair the governor valve control linkages

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Fire Area Tours

a. Inspection Scope

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

- April 7, 2005, Unit 1, Fire Zone SE018 - Train B switchgear Rooms 103 - 105

- April 7, 2005, Unit 1, Fire Zone SB004 - 790' safeguards building corridor Rooms 64, 70, and 71

- March 29 and April 15, 2005, Unit 2, Fire Area 2CA - containment building, all elevations

- June 6, 2005, Unit 1, Fire Zone SK017 - safeguards building, main steam and feedwater penetration areas, Rooms 100, 100a, 108, 109, and 110
- June 7, 2005, Unit 2, Fire Zone 2SK017 - safeguards building, main steam and feedwater penetration areas, Rooms 2-100, 2-100a, 2-108, 2-109, and 2-110
- June 16, 2005, Unit 2, Fire Zone 2SE018 - Train B switchgear Rooms 2-103 through 2-105

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill

a. Inspection Scope

The inspector observed a fire brigade drill on June 20, 2005, to evaluate the readiness of licensee personnel to prevent and fight fires, including the following aspects: (1) use of protective clothing, (2) use of breathing apparatuses, (3) placement and use of fire hoses, (4) entry into the fire area, (5) use of firefighting equipment, (6) brigade leader command and control, (7) communications between the fire brigade and control room, (8) searches for fire victims and fire propagation, (9) use of prefire plans, (10) adherence to the drill scenario, and (11) the drill critique. The licensee simulated a fire in the Unit 2 main transformer. The inspector completed one sample.

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

.1 External Flood Protection

a. Inspection Scope

The inspectors conducted an inspection of external 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 external sources. Based on that review and a review of the probabilistic risk assessment, a walkdown of the protected area was performed on May 2, 2005, to assess the adequacy of the flood protection measures following installation of the concrete vehicle barrier around the protected area perimeter. Interviews were conducted with the engineer responsible for the flooding calculation and the 10 CFR 50.59 screening for the vehicle barrier installation.

The inspector completed one sample.

b. Findings

No findings of significance were identified.

.2 Internal Flood Protection

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 probabilistic risk assessment, a walkdown of the Units 1 and 2 safeguards buildings, Rooms 78 (sampling room), 79, and 82 (corridor), on 810' elevation, was performed on June 6 and 8, 2005, to assess the adequacy of the flood protection measures. The walkdown included determining whether mitigating systems defined in the flood analysis were in place and functional. Interviews were conducted with the responsible engineer for the flooding calculation and a probabilistic risk analyst concerning the area in question.

The inspector completed one sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (71111.08P)

The procedure requires one sample of each activity described in Sections 02.01, 02.02, 02.03, and 02.04.

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

a. Inspection Scope

The inspector completed the required activity of Section 02.01 by observing and reviewing two types of nondestructive examination activities (i.e., volumetric examinations, and surface examinations); reviewing three recordable indications that were accepted for continued service; and reviewing two weld records for welding on Class 2 systems. The observation of these two types of examinations constitutes completion of the required sample.

The review and observation of nondestructive examinations were performed on the following components:

WELD NUMBER	WELD DESCRIPTION	EXAMINATION METHOD
8MT01	Feedwater Inlet Nozzle to Steam Generator 3 Weld	Magnetic Particle
8UT03A	Steam Generator 3 Circumferential Welds TCX-2-1100-3-2, -3, -9	Ultrasonic
8UT03B	Steam Generator 3 Circumferential Welds TCX-2-1100-3-2, -3, -9	Ultrasonic
8UT03C	Steam Generator 3 Circumferential Welds TCX-2-1100-3-2, -3, -9	Ultrasonic

The review of accepted recordable indications was for the following:

Evaluation of Recordable Indications on Hangers FW-2-017-702-C72R and FW-2-017-453-C62R and Loose HILTI Nuts	October 21, 2003
Evaluation of Damage to RPV Studs 33 thru 38	October 14, 2003
Evaluation of Loose Bolt on Pipe Clamp for Pipe Support SI-2-033-405-C41S	October 20, 2003

The weld records reviewed were:

Number	Title	Revision
CMWO 4-02-142566-00	Rework/Replace Valve 2FW-0230	0
DMWO 2-03-147427-00	Install New Piping	0

During the review of these examinations and evaluations, the inspector verified that the correct nondestructive examination procedure was used, examinations and conditions were as specified in the procedures, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspector also reviewed the documentation to determine if indications revealed were compared against the American Society of Mechanical Engineers (ASME) Code specified acceptance standards, and that the indications were appropriately dispositioned. The nondestructive examination certifications of those personnel observed performing examinations or identified during review of completed examination packages were reviewed by the inspector.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Upper Head Penetration Inspection Activities

The inspection requirements of Section 02.02 are only applicable after completion of Temporary Instruction 2515/150, "Reactor Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009)," Revision 3. The performance of the volumetric examination for this temporary instruction on Unit 2 is documented in Section 4OA5.3 of this report.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

Section 02.03 requires the review of a sample of boric acid corrosion control walkdown activities, a verification that visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components, a review of at least one engineering evaluation, and a review of at least one corrective action performed for identified boric acid leakage. The inspector performed the required reviews to confirm that the activities were performed in accordance with licensee procedures and ASME Code and regulatory requirements. This review meets the required sample for Section 02.03.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

Section 02.04 requires an assessment of in-situ pressure testing 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; a comparison of the predicted versus actual size and number of flaws detected; a confirmation that the scope of examinations and expansion criteria met the technical specification requirements, the EPRI Guidelines, and commitments made to the NRC; a verification that the licensee enveloped any new degradation mechanisms; a confirmation that all areas of potential degradation were inspected; a confirmation that any repairs were approved for use at the site; a confirmation that the plugging criteria were adhered to; an assessment of evaluation of any leakage >3 gallons per day; a confirmation that the probes and equipment were qualified; and a review of corrective actions for any loose parts identified during the inspections.

The inspector reviewed the in-situ pressure testing screening criteria to confirm that the criteria were in accordance with EPRI Guidelines, Technical Specifications, and commitments made to the NRC.

The inspector reviewed the licensee's report, "2RF07 Steam Generator Condition Monitoring and (Cycle 8) Operational Assessment," Revision 2. 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 inspector noted that the licensee engineers predicted that 4 tubes would require plugging during this outage. No tubes were required to be plugged; however, licensee personnel elected to plug 12 tubes as a preventive measure. The inspector noted that the projected growth of the indications in the 12 tubes would require plugging at the next refueling outage.

The inspection procedure specified confirmation be made that the steam generator tube eddy-current testing scope and expansion criteria met Technical Specification requirements, EPRI guidelines, and commitments made to the NRC. The inspector's review determined that the steam generator tube eddy-current testing scope and expansion criteria were being met.

The inspection procedure also specified that, if the licensee identified new degradation mechanisms, then verify that the licensee had fully enveloped the problem in an analysis and had taken appropriate corrective actions before plant startup. At the time of this inspection, no new degradation mechanisms had been identified.

The inspection procedure also required confirmation that all areas of potential degradation were being inspected, especially areas which were known to represent potential eddy-current testing 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 testing-challenged areas, were included in the scope of inspection and were being inspected.

The inspection procedure further required that repair processes being used were approved in the Technical Specifications for use at the site. The inspector verified that the repair criteria were in accordance with EPRI Guidelines, Technical Specifications, and commitments made to the NRC.

The inspection procedure required confirmation that the Technical Specification plugging limit was being adhered to and 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.

The inspection procedure stated that, if steam generator leakage >3 gallons per day was identified during operations or during postshutdown visual inspections of the tubesheet face, then assess whether the licensee had identified a reasonable cause and corrective actions for the leakage based on inspection results. The inspector did not conduct any assessment because this condition did not exist.

The inspection procedure required confirmation that the eddy-current testing 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 the eddy-current testing performed. During these examinations, the inspector 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 the review of one to five samples of eddy-current testing data if questions arose regarding the adequacy of eddy-current testing data analyses. The inspector did not identify any results where eddy-current testing data analyses adequacy was questionable.

The inspection performed met the requirements for the sample size of Section 02.04.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

c. Inspection Scope

The inspector reviewed eight inservice inspection-related corrective action documents (Smart Forms and associated evaluations) issued during the current and past refueling outages and verified that the licensee identified, evaluated, corrected, and trended problems. In this effort, the inspector evaluated the effectiveness of the licensee's corrective action process, including the adequacy of the technical resolutions.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Biennial Inspection (71111.11B)

a. Inspection Scope

The inspector evaluated the licensee's sample plan of the written examinations for compliance with 10 CFR 55.59 and NUREG-1021, as referenced in the facility requalification program procedures, and evaluated maintenance of license conditions for compliance with 10 CFR 55.53 by review of facility records (medical and administrative), procedures, and tracking systems for licensed operator training, qualification, and watch standing. In addition, the inspector reviewed remedial training for examination failures for compliance with facility procedures and responsiveness to address failed areas.

Furthermore, the inspector interviewed five personnel, including two operators, two instructors/evaluators, and an operations support person, regarding the policies and practices for administering requalification examinations. The inspector also reviewed operator performance on the written and operating examinations. Examination results were assessed to determine if they were consistent with the guidance contained in NUREG 1021 and Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process."

The review included an assessment of 25 operating examination job performance measures and 10 scenarios that were used in the biennial requalification cycle to determine if they provided adequate discrimination at the minimum acceptable level of operator performance.

The inspector also reviewed the remediation process and the results of the biennial written examination. The results of the examinations were assessed to determine the licensee's appraisal of operator performance and the feedback of performance analysis to the requalification training program. The inspector interviewed members of the training department and reviewed minutes of Program Review Board meetings to assess the responsiveness of the licensed operator requalification program.

Additionally, the inspector assessed the CPSES plant-referenced simulator for compliance with 10 CFR 55.46, Simulator Facilities, using Baseline Inspection Procedure 71111.11 (Section 03.11). This assessment included the adequacy of the licensee's simulation facility for use in operator licensing examinations and for satisfying experience requirements as prescribed by 10 CFR 55.46. The inspector reviewed a sample of simulator performance test records (transient tests, surveillance tests, and malfunction tests,) simulator action report records, and processes for ensuring simulator fidelity commensurate with 10 CFR 55.46. The inspector reviewed selected simulator action reports generated by the licensee that did not result in changes to the configuration of the simulator to assess the responsiveness of the licensee's simulator configuration management program. The inspector also interviewed members of the licensee's simulator configuration control group as part of this review.

In addition to the above, the inspector reviewed the licensee's extent of condition report for the failures in the initial operator licensing course to ascertain whether there were any linkages to the licensed requalification program. None were identified.

b. Findings

No findings of significance were identified.

.2 Quarterly Operator Requalification (71111.11Q)

a. Inspection Scope

The inspector observed a licensed operator requalification training scenario in the control room simulator on June 8, 2005. The training session began with a short lesson on immediate operator actions for a centrifugal charging pump trip. The simulator scenario consisted of: a loss of one source of offsite power and grid stability issues due to weather, Station Service Water Pump 1-02 trip, failure of a controlling steam flow channel, loss of all offsite power, loss of all station service water, and a subsequent recovery of one train of station service water and Emergency Diesel Generator (EDG) 1-02.

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.

b. Findings

No findings of significance were identified.

.3 Annual Reactor Operator Requalification Examination Integrity (71111.11Q)

a. Inspection Scope

The inspectors reviewed licensee requalification activities and corrective action documents pertaining to the annual requalification examination.

b. Findings

Introduction. A Green, self-revealing, noncited violation (NCV) was identified for the failure to protect the integrity of the annual reactor operator requalification examination as described in 10 CFR 55.49.

Description On February 16, 2005, the simulator was being prepared to be turned over to the initial operator licensing class following the administration of an annual reactor operator requalification simulator examination. The licensee was in week 2 of a 6-week examination cycle. Completion of the examination was approximately 2 hours behind schedule. During preparation of the simulator by the oncoming initial operator simulator instructors, annual requalification examination material was discovered in the simulator booth. The duration that the examination material was left unattended was short.

Analysis. A compromise in the annual reactor operator requalification examination integrity has the potential to allow unqualified operators to operate the reactor. Therefore, the finding was determined to be more than minor because, if left uncorrected, the finding could become a more significant safety concern. Based on the results of a significance determination process (SDP) using Manual Chapter 0609, Appendix I, this finding was determined to have very low safety significance, since the licensee took immediate compensatory measures, which included destroying the examination material and modifying the examinations for the subsequent operating crews. The inspector determined the cause of the violation was related to the personnel aspect of the human performance crosscutting area.

Enforcement. 10 CFR 55.49 states that applicants, licensees, and facility licensees shall not engage in any activities that compromise the integrity of any application, test, or examination. The integrity of an examination is considered compromised if any activity regardless of intent affected or, because of detection, would have affected the equitable and consistent administration of the test or examination. The failure to control annual requalification examination material on February 16, 2005, constitutes a violation of 10 CFR 55.49. Because the violation was determined to be of very low safety significance and it was entered into their corrective action program as Smart Form (SMF) SMF-2005-0615, this violation is being treated as an NCV, consistent with Section IV.A of the NRC Enforcement Policy (NCV 05000445, 446/2005003-01).

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:

- Diesel Driven Fire Pump X-05 was placed in a(1) status due to repetitive maintenance preventable functional failures from July 2004 through January 2005. This issue was entered into the corrective action program as Smart Forms SMF-2004-2527 and SMF-2005-0105.

- Reviewed the performance history of the Unit 2 turbine-driven auxiliary feedwater pump from September 2004 to May 2005, including repair and postmaintenance testing for various equipment issues on May 12-20, 2005: Quick Technical Evaluation QTE-2005-2054-01 for inability to reduce turbine speed to minimum, SMF-2005-2130 for governor valve not fully opening, and SMF-2005-2137 for loss of speed indication.

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

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed 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:

- March 30, 2005, Unit 1, scheduled maintenance and surveillance testing of the EDG 1-01 during the 345 kV west bus outage

- April 13, 2005, Unit 1, scheduled maintenance and surveillance testing of the EDG 1-02 while performing maintenance work in the 345 kV switchyard on the Parker Substation Feeder Breaker 8040

- May 19, 2005, Unit 2, emergent work to repair the governor valve control linkages on the turbine-driven auxiliary feedwater pump concurrent with scheduled EDG surveillance testing

- June 5-8, 2005, Unit 2, emergent work on Component Cooling Water Pump 2-02 and its associated recirculation Valve 2-FV-4537 when the valve was discovered to have lost part of its rubber liner

- June 13-16, 2005, Unit 1, emergent work on Atmospheric Relief Valve 1-PV-2326 for Steam Generator 1-02 when the booster valve was found leaking during scheduled maintenance

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

For the two nonroutine events described below, the inspectors observed the simulator just-in-time training and reviewed the applicable procedures prior to the evolution. The inspectors attended prejob briefings and observed portions of the evolution from the control room. Procedural use, communications, coordination between organizations, and safe operation of the plant during the evolution were evaluated to ensure risk was minimized and safety was maintained.

- On March 26, 2005, the control room operators commenced the Unit 2 reactor shutdown to begin Refueling Outage 2RF08 via boration as per Integrated Plant Operating (IPO) IPO-003B, "Power Operations," Revision 14. At 11:46 am, with the reactor at approximately 16 percent power, the operators manually tripped the reactor and entered EOP 0.0B, "Reactor Trip or Safety Injection," Revision 1. Operators transitioned to EOS-0.1B, "Reactor Trip Response," Revision 1 and IPO-005B, "Plant Cooldown From Hot Standby to Cold Shutdown," Revision 7. The inspectors observed control room activities and operator actions during the evolution to ensure formal and clear communications, proper procedure usage, command and control activities, proper use of emergency procedures, and the controlled and safe shutdown of the Unit 2 reactor.

- On April 22, 2005, the control room operators lowered Unit 2 reactor coolant system water level to 57 inches above the reactor core (midloop) in preparation to remove steam generator nozzle dams and install steam generator primary manways. The inspectors reviewed Generic Letter 88-17, "Loss of Decay Heat Removal" and TXU's responses. Procedure IPO-010B, "Reactor Coolant System Reduced Inventory Operations," Revision 9, was reviewed to ensure adequate controls were in place. The control room activities and operator's actions were observed during the evolution to ensure the procedure was followed, plant instruments were responding correctly, conservative decisions were made, and that the evolution was completed safely. Control room activities were periodically observed for distractions to the operators while the reactor vessel water level remained in reduced inventory.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- Evaluation (EVAL) EVAL-2005-1098-01-01, to determine the operability of Unit 1 Containment Spray Chemical Eductor 1-04 due to the discovery of a missing bolt from the flange connection to the containment spray discharge piping, reviewed on March 28, 2005

- Final Design Authorization (FDA) FDA-2005-1483-01-01, including a determination of the operability of the Unit 2 pressurizer with Pressurizer Heaters BUA19, BUA20, and BUA21 de-energized for the next operating cycle, reviewed on May 5, 2005

- EVAL-2002-2566-06-00, to confirm the operability of Units 1 and 2 EDGs 1EG1, 1EG2, 2EG1, and 2EG2 with administrative controls in effect during surveillances with the EDG connected to offsite power, reviewed on May 6, 2005

- EVAL-2005-1689-01-00 and -01, to determine the operability of Unit 2 Station Service Water Pump 2-02. The upper motor bearing associated with this pump showed signs of potential degradation when the motor was replaced during Refueling Outage 2RF08.

- EVAL- 2005-1588-01-00, to determine the operability of 2-HV-4574-MO, Containment Spray Heat Exchanger 2-01 component cooling water return valve motor operator, after the discovery that the maximum torque was in excess of the procedural limit of 2200 ft-lbs, reviewed on June 6, 2005

- Quick Technical Evaluation QTE- 2005-2054-01-00, to determine operability of the Unit 2 turbine-driven auxiliary feedwater pump following an inability to reduce turbine speed to below 2400 rpm, reviewed on May 12 and June 6, 2005

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Selected Operator Workarounds

a. Inspection Scope

On May 4, 2005, the inspectors reviewed the compensatory action implemented for Unit 1 EDG 1-01 due to a failure of the annunciator power supply to determine the impact on operations and the ability to implement abnormal or emergency operating procedures. A control room supervisor was interviewed, alarm procedures, Technical Specifications, and Smart Form SMF-2005-1943-00 were reviewed.

The inspector completed one sample.

b. Findings

No findings of significance were identified.

.2 Cumulative Review of the Effects of Operator Workarounds

a. Inspection Scope

On May 4, 2005, the inspector reviewed the cumulative effects of operator workarounds to determine: (1) the reliability, availability, and potential for misoperation of a system; (2) if multiple mitigating systems could be affected; (3) the ability of operators to respond in a correct and timely manner to plant transients and accidents; and (4) if the licensee has identified and implemented appropriate corrective actions associated with operator workarounds. The inspector completed one sample.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

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

- April 10, 2005, Unit 2, Train B station service water motor replacement per Work Order (WO) WO-1-04-156847-00, tested in accordance with Operability (OPT) OPT-207B, “Service Water System,” Revision 11

- April 21, 2005, Unit 2, Train A safety injection motor and pump replacement per WO-3-01-327783-01 and Train A solid state safeguards sequencer driver card replacement per WO-4-04-153883-00, tested in accordance with OPT-430B, “Train A Integrated Test Sequence,” Revision 5

- April 22, 2005, Unit 2, Train A EDG mechanical preventive maintenance per WO-3-03-336977-01, tested in accordance with OPT-214B, “Diesel Generator Operability Test,” Revision 12

- April 23, 2005, Unit 2, Penetration 2-MII-0009 opened for outage communications, local leak rate test hoses, and instrument air lines per WO-3-03-339628-01, tested in accordance with OPT-813B, “Appendix J Leak Rate Test of Penetration 2-MII-0009,” Revision 2

- April 25, 2005, Unit 2, Train B station service water motor replacement per WO-1-05-161265-00 and SMF-2005-1689-00, tested in accordance with OPT-207B, “Service Water System,” Revision 11

- May 19, 2005, Unit 2, turbine-driven auxiliary feedwater pump governor valve linkage repair and adjustment in accordance with WO-4-05-161809-00, approved troubleshooting plan for the governor valve linkage, tested in accordance with OPT-206B, “AFW System,” Revision 17

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

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated the licensee’s Refueling Outage 2RF08 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 inspections performed and outage activities reviewed and/or observed by the inspectors included:

- Discussions and review with the Outage Manager of the outage schedule concerning risk
- Unit shutdown and cooldown
- Containment walkdowns to identify indications of reactor coolant leakage
- Reduced inventory and midloop activities to perform steam generator nozzle dam removal and manway installation
- Reactor coolant system instrumentation, including Mansell level instrumentation
- Defense in depth and mitigation strategy implementation
- Containment closure capability
- Verification of decay heat removal system capability
- Spent fuel pool cooling capability
- Reactor water inventory control, including flow paths, configurations, alternate means for inventory addition, and controls to prevent inventory loss
- Controls over activities that could affect reactivity
- Refueling activities that included fuel offloading, fuel transfer, and core reloading
- 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; system alarm an annunciator functionality; and the effectiveness of the licensee's problem identification and correction program. The following six surveillance test activities were observed and/or reviewed by the inspectors:

- April 22, 2005, Unit 2, Train A EDG 24-hour load run in accordance with OPT-214B, "Diesel Generator Operability Test," Revision 12

- April 22, 2005, Unit 2, containment recirculation sump inspection in accordance with OPT-306, "Containment Sump Inspection," Revision 6

- April 24, 2005, Unit 2, local leak rate test on the containment equipment hatch seal in accordance with OPT-805B, "Appendix J Leak Rate Test of Equipment hatch Seal," Revision 1

- April 26, 2005, Unit 2, control rod drop time test in accordance with Nuclear Engineering Procedure NUC -206, "Control Rod Drop Timing (Plant Computer Method)," Revision 14, in conjunction with OPT-117, "Digital Rod Position Indication System," Revision 2 and INC-3026, "Control Rod Drive Mechanism (CRDM) Step Traces," Revision 1

- April 27, 2005, Unit 2, low power physics testing in accordance with Nuclear Engineering Procedure NUC-301, "Low Power Physics Testing,"Revision 11

- April 28, 2005, Unit 1, Train A EDG operability test in accordance with OPT-214A, "Diesel Generator Operability Test," Revision 18

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

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

- Installation of the alternate power diesel generators in accordance with Maintenance Section - Electrical Procedure MSE-G2-0850, "Unit 2 Alternate Power Diesel Generators Installation and Removal," Revision 1.

- Installation of a Rosemount 1151DP3 transmitter in place of an ITT Barton 752 transmitter to monitor the Safety Injection Accumulator 2-04 level as Level Transmitter 2-LT-0957 in accordance with SMF-2005-1644 and FDA-2005-1644.

- The application of a seal weld on residual heat removal to cold leg injection Check Valves 2-8818B and 8818C in accordance with WO-4-05-160932 and WO-4-05-160806-00, respectively, to repair a body to bonnet leak.

The inspectors completed three samples.

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, engineering controls, and air sampler locations

- Conformity of electronic personal dosimeter alarm setpoints with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms.

- Barrier integrity and performance of engineering controls in two airborne radioactivity areas

- Physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools.

- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection

- Corrective action documents related to access controls

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls such as, required surveys, radiation protection job coverage, and contamination controls during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to 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:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem CEDE

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

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

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspector sampled licensee submittals for the performance indicators listed below from April 2004 to April 2005. To verify the accuracy of the performance indicator data reported during that period, performance indicator definitions and guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis in reporting for each data element.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness Performance Indicator

Licensee records reviewed included corrective action documentation that identified occurrences of locked high radiation areas (as defined in the licensee’s Technical Specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in Nuclear Energy Institute 99-02). Additional records reviewed included ALARA (as low as is reasonably achievable) 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.

Public Radiation Safety Cornerstone

- 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 inspectors 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 Semiannual Trend Review

a. Inspection Scope

On June 22, 2005, the inspectors completed a semiannual review of licensee internal documents, reports, performance indicators and an audit to identify trends that might indicate the existence of more safety significant issues. The inspectors reviewed the following types of documents:

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

b. Findings and Observations

No findings of significance were identified. However, during the review, the inspectors did note the following three items: (1) a decline in human performance was noted by TXU in operations activities in support of the Unit 2 refueling outage; (2) a continuing trend of failures of time delay relays; and (3) TXU recognized the continuing need to improve the corrective action program. The inspectors did not identify any additional trends.

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

.2 Daily Condition Report Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing the licensee's computerized corrective action program database (SMFs), reviewing hard copies of selected SMFs and attending related meetings such as Plant Event Review Committee meetings.

b. Findings

No findings of significance were identified.

.3 Section 2OS1 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding access controls to radiologically significant areas and radiation worker practices. The inspector reviewed corrective action documents for root cause/apparent cause analysis against the licensee's problem identification and resolution process. No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 Operator de-energized Battery Charger BC2D2 instead of Battery Charger BC2D3

a. Inspection Scope

The inspector reviewed SMF-2005-001652-00, describing an event on April 15, 2005, in which a plant operator inadvertently de-energized Battery Charger BC2D2 instead of Battery Charger BC2D3. Also reviewed was the Plant Event Review Committee meeting minutes and System Operating Procedure SOP-606B "24/48V & 125/250 VDC Switchgear and Distribution Systems, Batteries & Battery Chargers," Revision 3.

b. Findings

Introduction. A Green, self-revealing, NCV for failure to follow procedure was identified when a plant operator inadvertently de-energized Unit 2 Battery Charger BC2D2 instead of the assigned action of de-energizing Battery Charger BC2D3.

Description. On April 15, 2005, a plant operator was assigned to de-energize Unit 2 Battery Charger BC2D3 for scheduled maintenance. Using SOP-606B, the operator went to the wrong battery charger and opened the circuit breakers on Battery Charger BC2D2. Unit 2 was in a scheduled refueling outage with Train A bus work in progress. In this configuration, the only power source for the Unit 2 control room annunciators was Battery Charger BC2D2. As a result of de-energizing Battery Charger BC2D2, all of the Unit 2 main control board annunciators were lost, which potentially affected plant operators' emergency assessment capability. In addition, Unit 2 containment ventilation isolation occurred and some turbine building drains were diverted.

Immediate actions taken following the inadvertent de-energizing of Battery Charger BC2D2 included restoring power to dc Bus 2D2 by placing Battery Charger BC2D2 back in service and entering ABN-740B for the loss of control room annunciation. The total time that Battery Charger BC2D2 was de-energized was approximately 7 minutes.

Unit 2 was in a shutdown condition in Mode 6 at the time of the event. Core reload had been completed and no core alterations were in progress. Reactor refueling cavity level was greater than 23 feet, cavity level indication was maintained throughout the event by means of the installed Mansell instruments, and the plant computer remained available. Train B residual heat removal continued to operate in the shutdown cooling mode and Train B component cooling water, station service water, and safety chill water continued to operate.

Analysis. The performance deficiency associated with this violation was failure of CPSES personnel to follow SOP-606B to de-energize the correct battery charger. This finding is greater than minor because it affected the configuration control, equipment performance, and human performance attributes of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. The finding was processed through the Inspection Manual Chapter 0609 Appendix G shutdown operations SDP, using Table 1 and Checklist 4, "PWR Refueling Operation: RCS Level >23' or PWR Shutdown Operation with time to boil >2 hours and Inventory in the Pressurizer," and determined to be of very low safety significance (Green). It was determined that Section III, "Power Availability Guidelines," of Checklist 4 was effected by this event in that the requirements of Technical Specification 3.8.5, "DC Sources - Shutdown," were not met during the event but the finding did not: (1) increase the likelihood of a loss of reactor coolant system (RCS) inventory, (2) degrade the licensee's ability to terminate a leak path or add RCS inventory when needed, nor (3) degrade the licensee's ability to recover decay heat removal once it is lost. The inspector determined the cause of the violation was related to the personnel aspect of the human performance crosscutting area.

Enforcement. Technical Specification 5.4.1 requires that written procedures be implemented covering the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Section 3.s.(2)(c), specifies procedures covering the operation of the dc electrical system. On April 15, 2005, plant operators failed to implement required procedures when the wrong battery charger was de-energized, resulting in the loss of all Unit 2 control room annunciators. Because this violation was of very low safety significance and it was entered into the corrective action program as SMF-2005-001652-00, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000446/2005003-02).

.2 (Closed) Licensee Event Report (LER) 05000445,446/2004-002-00 Missed Surveillance on Loss of Power Emergency Diesel Generator Start Instrumentation

On April 2, 2004, the licensee identified that not all components of the undervoltage channels were being tested to meet channel calibration requirements at a frequency of 18 months as specified in Technical Specification Surveillance Request 3.3.5.3. The licensee has determined that the cause of the missed surveillance was a less than adequate review of a change to the preventive maintenance database frequency for these Agastat timing relays due to personnel errors in the review process and inadequate procedure referencing. All seven affected functions on Units 1 and 2 were verified successfully via the required Technical Specification channel calibrations during their respective refueling outages. Corrective actions taken by the licensee included: a lessons learned concerning the event was issued to organizations involved, revision of Technical Specification training modules concerning “channel” and “channel calibrations,” a review of past test frequencies for electrical components to ensure compliance with current Technical Specification, and a revision to the corrective action program procedure to refer any question involving interpretation of Technical Specification requirements to regulatory affairs for resolution. No new findings were identified by 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 has documented this issue in SMF-2004-1177-00. This LER is closed.

4OA4 Crosscutting Aspects of Findings

.1 The finding described in Section 1R11.3 of this report was related to the personnel aspect of the human performance crosscutting area, in that the instructors did not maintain adequate control of the annual requalification examination material.

.2 The finding described in Section 4OA3.1 of this report was related to the personnel aspect of the human performance crosscutting area, in that the operator failed to perform an adequate self-check.

4OA5 Other Activities

.1 Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01) (Temporary Instruction 2515/160)

This Temporary Instruction provided the guidelines to verify compliance with licensee commitments to NRC Bulletin 2004-01, “Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors.” The inspector used the inspection requirements for the bare metal visual examination to conduct this inspection on the CPSES Unit 2 pressurizer and steam space penetrations during Refueling Outage 2RF08, Spring 2005.

a. Inspection Scope

The inspector performed this performance-based evaluation and assessment to ensure that the NRC had an independent review of the condition of the pressurizer and steam space piping alloy 82/182 dissimilar metal welds. The inspector assessed the effectiveness of the licensee examinations of the pressurizer vessel and penetrations. Specifically, the inspector:

- Met with licensee representatives to review and discuss inspections plans and contingencies
- Attended prejob briefs
- Directly inspected and assessed the condition of the pressurizer and the associated piping weld penetrations

- Assessed the physical difficulties in performing the inspection, which included any debris, dirt, boron, and other viewing impediments
- Interviewed the licensee inspectors
- Assessed the licensee’s ability to distinguish small boron deposits located at the weld locations
- Reviewed completed records, including the final engineering inspection report for CPSES Unit 2
- 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 inspector also reviewed the following documents during this inspection:

- NRC Bulletin 2004-01, “Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors,” dated May 28, 2004
- NRC Information Notice 2004-11, “Cracking in Pressurizer Safety and Relief Nozzles and in Surge Line Nozzle,” dated May 6, 2004
- CPSES 60-Day response to NRC Bulletin 2004-01, “Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at pressurized Water Reactors,” TXX-04140, dated July 27, 2004
- CPSES response to NRC’s Request for Additional Information regarding the response to NRC Bulletin 2004-01, “Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at pressurized Water Reactors,” TXX-05056, dated March 7, 2005
- CPSES Station Administration Manual Procedure STA-737, “Boric Acid Corrosion Detection and Evaluation,” Revision 3
- NRC Inspection Manual, Inspection Procedure 57050, “Visual Testing Examination,” issued March 9, 1999

b. Findings

No findings of significance were identified. The inspector concluded that the licensee met the applicable commitments in that they performed a 100 percent bare metal visual inspection of the circumference over the axial length of the Alloy 82/182 identified welds for the Unit 2 pressurizer. These inspections were performed by a VT-2 Level II certified examiner. The inspector has provided the following details of the inspection as required by Temporary Instruction 2515/160, "Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)," issued October 6, 2004.

1. Examination

   The licensee's examiner was certified in accordance with CPSES procedures to meet ASME Section XI for VT-2 Level II.

   The examination was conducted in accordance with a CPSES examination plan, "RCS Pressure Boundary DM Weld Supplemental Visual Examination Plan," Revision 1, approved on March 28, 2005. The examination plan provided: (1) responsibilities for the examination process; (2) examiner qualification; (3) scope of welds to be examined, a description of the basic bare metal inspection technique and the expectation of 100 percent inspection coverage; (4) acceptance criteria for the inspection; (5) types of indications that shall be further investigated; (6) criteria for cleaning the examined area; and (7) sufficient guidance to satisfy licensee commitments for the inspection. The inspectors concluded that the inspection plan, combined with training, have provided adequate guidance for the licensee examiner to identify, disposition, and resolve deficiencies.

   Due to the proximity of the bare metal visual examination, VT-2 Level II qualified personnel, and the accessability of the specified Alloy 82/182 welds, the inspectors determined that RCS leakage described in NRC Bulletin 2004-01 would be identified, if present.

2. Physical condition penetration nozzles and steam space piping

   In general, the condition of the weld areas examined were in excellent condition. Access to the welds only required the removal of a relatively small amount of mirror insulation, radiation levels were acceptable, and the welds themselves were very new looking with no residue of previous spills or in-service inspections. Only on the downhill side of the spray, safety, and pressurizer power-operated relief valve welds was it necessary to use a mirror (due to limited space below the piping). All other examinations were performed with the naked eye.

3. Visual inspection protocol

   Direct visual inspection and the use of a mirror were the inspection techniques used by qualified examination personnel.

4. Inspection coverage

   The inspectors observed that the licensee completed a 100 percent, 360 degree bare metal inspection of the pressurizer penetration nozzles and steam space piping connections.

5. Capability to identify and characterize small boric acid deposits

   The inspectors determined that the direct visual inspections, coupled with mirror assisted visual inspections, were capable of detecting, identifying, and characterizing small boric acid deposits, if present, as described in NRC Bulletin 2004-01. This fact was determined via direct inspection during the licensee inspection of the pressurizer and associated steam space piping connections.

6. Identified deficiencies that required repair

   No deficiencies were identified.

7. Impediments to effective examinations

   There were no impediments that adversely affected effective bare metal visual examinations. In all examination cases, mirror insulation was required to be removed. The examination of the pressurizer safety and power-operated relief valve line welds was supplemented by a mirror to allow examination of the downhill side of the welds. The dose rates were acceptable, and the inspectors received approximately 50 mRem to complete the in-plant portion of the temporary instruction.

8. Techniques used for augmented inspections

   Augmented inspections were not required.

9. Appropriateness of follow-on examinations

   Follow-on examinations were not required.

.2 <u>Transportation of Reactor Control Rod Drives in Type A Packages (Temporary Instruction 2515/161)</u>

a. <u>Inspection Scope</u>

This area was inspected to verify that the licensee's radioactive material transportation program complies with specific requirements of 10 CFR Parts 20 and 71 and Department of Transportation regulations contained in 49 CFR Part 173. The inspector interviewed licensee personnel and determined the licensee had undergone refueling/defueling activities between January 1, 2002, and present, but it had not shipped irradiated control rod drives in Department of Transportation Specification 7A Type A packages.

b. <u>Findings and Observations</u>

No findings of significance were identified.

.3 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, Revision 3 (Temporary Instruction 2515/150)

a. Inspection Scope

The inspector observed and reviewed licensee activities associated with the reactor pressure vessel head and vessel head penetration nozzle inspection that were implemented in accordance with the requirements of Order EA-03-009. The review was performed in accordance with Section 03.03, “Volumetric Examination,” of Inspection Procedure 71111.08.

The licensee performed ultrasonic and eddy-current examinations of all control element drive mechanism penetrations and the head vent penetration. The licensee did not perform examinations of the two reactor vessel level instrumentation system penetrations and rescheduled these examinations for the next refueling outage. The inspector independently reviewed the inspection results for eight of the penetrations. The licensee did not identify any nozzle or weld degradation.

NRC inspectors have performed the bare metal head inspections on Unit 1 during the fall refueling outage in 2002 (NRC Inspection Report 05000445; 446/2002-005) and on Unit 2 during the fall 2003 refueling outage (NRC Inspection Report 05000445; 446/2004-002). The Unit 1 bare metal inspection was performed using Temporary Instruction 2515/145, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." The Unit 2 bare metal inspection was performed using Temporary Instruction 2515/150.

The licensee plans to replace the Unit 1 reactor vessel head prior to the end date of the temporary instruction. Because the surface and volumetric examinations have been completed on Unit 2, the surface examination is complete on Unit 1, and the Unit 1 reactor vessel head will be replaced prior to February 11, 2009, this temporary instruction is completed and may be closed.

b. Findings

No findings of significance were identified.

.4 Operational Readiness of Offsite Power (Temporary Instruction 2515/163)

The inspectors collected data pursuant to Temporary Instruction 2515/163, “Operational Readiness of Offsite Power.” The inspectors reviewed the licensee’s procedures related to General Design Criteria 17, “Electric Power Systems”; 10 CFR 50.63, “Loss of All Alternating Current Power”; 10 CFR 50.65(a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”; and the Technical Specifications for the offsite power system. The data was provided to the Office of Nuclear Reactor Regulation for further review. Documents reviewed for this temporary instruction are listed in the attachment.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 8, 2005, the inspector presented the results of the Inservice Inspection and Temporary Instruction 2515/150: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles inspection to Mr. S. Smith, Director, System Engineering, and other members of licensee management. Licensee management acknowledged the inspection findings. The inspector reviewed proprietary information during the inspection; however, no proprietary information is contained in the report.

On April 16, 2005, the inspector presented the results of the Access Control to Radiologically Significant Areas inspection to Mr. R. Flores, Vice President, Operations, and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On May 10, 2005, the inspector discussed the results of Temporary Instruction 2515/161, Transportation of Reactor Control Rod Drives in Type A Packages, inspection with Mr. J. Curtis, Radiation Protection Manager. The inspector verified that no proprietary information was provided during the inspection.

On June 3, 2005, the inspector briefed Mr. S. Sewell, Nuclear Training Manager, and other members of the licensee's management of the results of the biennial licensed operator requalification inspection. The licensee acknowledged the findings presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

The inspector presented the resident inspection results to Mr. M. Blevins, Senior Vice President and Chief Nuclear Officer, and other members of licensee management on June 29, 2005. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

On July 19, 2005, the inspector conducted a telephone exit meeting with Mr. F. Madden, Director, Regulatory Affairs, during which changes to the content of the inspection report were identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

O. Bhatty, Inservice Test Engineer
M. Blevins, Senior Vice President and Chief Nuclear Officer
S. Bradley, Supervisor, Health Physics, Radiation Protection & Safety Services
J. Curtis, Radiation Protection Manager, Radiation and Industrial Safety
D. Ellis, Level III Qualified Data Analyst
R. Flores, Vice President, Nuclear Operations
T. Hope, Manager, Regulatory Performance
R. Kidwell, Licensing Engineer
F. Madden, Director, Regulatory Affairs
P. Passalugo, Inservice Inspection Coordinator
P. Polefrone, Plant Manger
S. Sewell, Nuclear Training Manager
S. Smith, Director, System Engineering
T. Weyandt, Steam Generator Coordinator
D. Wilder, Radiation and Industrial Safety Manager

Other Personnel

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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000445,446/2005003-01	NCV	Failure to protect the integrity of the annual reactor operator requalification examination as described in 10 CFR 55.49 (Section 1R11.3)
05000446/2005003-02	NCV	Failure to follow procedure by de-energizing the wrong battery charger, resulting in the loss of all Unit 2 control room annunciators (Section 4OA3.1)

Closed

05000445,446/2004-002	LER	Missed Surveillance on Loss of Power Emergency Diesel Generator Start Instrumentation (Section 4OA3.2)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R08, Inserviсe Inspection:

Evaluations

EVAL-2003-003575-01-00
EVAL-2004-000816-01-00

Miscellaneous

NUMBER	TITLE	REVISION/ DATE
	Evaluation of Recordable Indications on Hangers FW-2-017-702-C72R and FW-2-017-453-C62R and Loose HILTI Nuts	October 21, 2003
	Evaluation of Damage to RPV Studs 33 thru 38	October 14, 2003
	Evaluation of Loose Bolt on Pipe Clamp for Pipe Support SI-2-033-405-C41S	October 20, 2003
8MT01	Magnetic Particle Examination Data Sheet for Feedwater Inlet Nozzle to Steam Generator Weld	April 1, 2005
8UT03A	0E Scan of Steam Generator 3 Circumferential Welds TCX-2-1100-3-2, -3, -9	April 1, 2005
8UT03B	45E Scan of Steam Generator 3Circumferential Welds TCX-2-1100-3-2, -3, -9	April 1, 2005
8UT03C	60E Scan of Steam Generator 3Circumferential Welds TCX-2-1100-3-2, -3, -9	April 1, 2005
27PT-01	Surface Examination Data RHR HX 1	September 18, 2003
27UT52	Calibration Data Sheet RV Closure Head Studs	October 15, 2003
27UT54	Calibration Data Sheet RV Closure Head Studs	October 17, 2003

Miscellaneous

NUMBER	TITLE	REVISION/ DATE
27VT55	Visual Examination Data Safety Injection	October 13, 2003
27VT59	Visual Examination Data Feedwater	October 15, 2003
27VT65	Visual Examination Data Safety Injection	October 15, 2003
27VT65	Visual Examination Data RV Studs	October 15, 2003
27VT66	Visual Examination Data feedwater	October 13, 2003
27VT68	Visual Examination Data RV Studs, Nuts and Washers	October 17, 2003
CMWO 4-02-142566-00	Rework/Replace Valve 2FW-0230	0
DMWO 2-03-147427-00	Install New Piping	0
FDA-2003-003111-01-00	Final Design Authorization for RPV Stud Repairs	November 6, 2003
LTR-SGDA-04-126	540 and 520 Groove Bobbin Probe Evaluation	April 18, 2004
STA-733	Steam Generator Reliability Program	8
WDI-LTR-05-003	NDE Level III Certification Letter	0, 1, 2, & 3
WPT-16505	2RF07 Steam Generator Condition Monitoring and (Cycle 8) Operational Assessment, Revision 2	February 2, 2004

Procedures

NUMBER	TITLE	REVISION
WPS CP-201	Welding Procedure Specification Manual GTAW, SMAW P1 Groups 1 & 2 (all combinations)	10
WPS CP-301	Welding Procedure Specification Manual GTAW, SMAW P8-P8	11

Procedures

NUMBER	TITLE	REVISION
TX-ISI-210	Ultrasonic Examination Procedure for Welds in Ferritic Steel Vessels	5
TX-ISI-212	Ultrasonic Examination Procedure of Nozzle Inner Radius Sections for CPSES	6
IPO-003A	Power Operations	22

Smart Forms

SMF-2003-003575-00
SMF-2004-000816-00
SMF-2005-000934-00
SMF-2005-000953-00
SMF-2005-000967-00
SMF-2005-001233-00

Section 1R11.1: Biennial Licensed Operator Requalification

CPSES Biennial Exam For 2003 & 2004 LORT Results
Procedure TRA-2004; "Licensed Operator Requalification Training" Revision 13
Licensed Operator Requalification Sample Plan 2003-2004
LORT 2005 Operational Exam; Revision 3/7/05
LORT 2005 Operational Exam; Revision 3/2005
Licensed Operator/STA Requal Curriculum (03/04)
LORT Biennial RO Written Exam LO49.G04.E2P
LORT Biennial Simulator Evaluation LO49.G04.E6P
LORT Biennial JPM Evaluation LO49.G04.E5P
LORT Biennial SRO Written Exam LO49.G04.E1P
LORT Biennial JPM Evaluation LO49.G04.E51
LORT Biennial Simulator Evaluation LO49.G04.E41
LORT Biennial RO Written Exam LO49.G04.E21
LORT Biennial SRO Written Exam LO49.G04.E11
LORT Biennial JPM Evaluation LO49.G04.E52
LORT Biennial Simulator Evaluation LO49.G04.E42
LORT Biennial RO Written Exam LO49.G04.E22
LORT Biennial SRO Written Exam LO49.G04.E12
LORT Biennial JPM Evaluation LO49.G04.E53
LORT Biennial Simulator Evaluation LO49.G04.E43
LORT Biennial RO Written Exam LO49.G04.E23
LORT Biennial SRO Written Exam LO49.G04.E13
LORT Biennial JPM Evaluation LO49.G04.E55
LORT Biennial Simulator Evaluation LO49.G04.E45
LORT Biennial RO Written Exam LO49.G04.E25
LORT Biennial SRO Written Exam LO49.G04.E15
LORT Biennial Simulator Evaluation LO49.G04.E46

LORT Biennial JPM Evaluation LO49.G04.E54
LORT Biennial Simulator Evaluation LO49.G04.E44
LORT Biennial RO Written Exam LO49.G04.E24
LORT Biennial Remedial Simulator Evaluation LO69.G04.E44
LORT Biennial SRO Written Exam LO49.G04.E14
Simulator Configuration Management, SOMI-009, Revision 7
Simulator Testing Program, SOMI-10, Revision 14
Training Load Promotion, Load Version T.0013; 4/29/2005
List of Open Simulator Action Requests
List of Closed Simulator Action Requests for 12 Months prior to 5/1/05
Root Cause Analysis for Initial License Class 15 Poor Performances
PS Training Feedback Report 04-1
OPS Training Feedback Report 04-6

Simulator Action Requests:
02SA0022; 03SA0149; 03SA0398; 04SA0322; 05SA0012; 05SA0061; 04SA0351; 04SA0073; 02SA0431; 02SA0288; 02SA0202; 01SA0015; 04SA0132

Section 2OS1: Access Control to Radiologically Significant Areas

Procedures

STA-650 General Health Physics Plan, Revision 5
STA-656 Radiation Work Control, Revision 12
STA-660 Control of High Radiation Areas, Revision 9
RPI-110 Radiation Protection Shift Activities, Revision 8
RPI-528 Multiple Dosimetry Badging, Revision 8
RPI-602 Radiological Surveillance and Posting, Revision 23
RPI-606 Radiation Work and General Access Permits, Revision 12

Radiation Work Permits

2003-2400 Primary-side Steam Generator Activities (2RF08)
2005-2404 Head Stand Area Volumetric Exam of Reactor Vessel Head
2004-1600 Refueling (1RF10)
2005-2600 Refueling (2RF08)

Corrective Action Documents (Smart Forms)

2004-3306, 2004-3501, 2005-0975, 2005-1276, 2005-1422, 2005-1593, 2005-1650

Audits and Self-Assessments

CPSES Nuclear Overview Department Evaluation Reports: 2004-016

Section 4OA2: Problem Identification and Resolution

Smart Forms

Related to time delay relay failures: SMF-2005-0425-00, SMF-2005-1139-00, SMF-2005-1152-00, SMF-2005-1787-00, SMF-2005-2228-00, and SMF-2005-2487-00

LIST OF ACRONYMS

ABN	Abnormal Conditions Procedure
ASME	American Society of Mechanical Engineers
CFR	*Code of Federal Regulations*
CPSES	Comanche Peak Steam Electric Station
CRDM	control rod drive mechanism
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
EVAL	evaluation
FDA	final design authorization
NCV	noncited violation
NRC	Nuclear Regulatory Commission
OPT	operability test
RCS	reactor coolant system
SMF	Smart Form
SOP	system operating procedure
SSC	structures, systems, or components
STA	station administrative procedure
WO	work order