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

December 27, 2004

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 and 2 - INSPECTION REPORT 05000445/2004008; 05000446/2004008

Dear Mr. Blevins:

On December 3, 2004, the Nuclear Regulatory Commission (NRC) completed an inspection at the Comanche Peak Steam Electric Station, Units 1 and 2. The enclosed report documents the inspection findings, which were discussed on December 3, 2004, with Mr. R. Flores, Vice President Nuclear Operations, and other members of your staff.

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

Based on the results of this inspection, the NRC has identified two violations of regulatory requirements that were evaluated using the Significance Determination Process described in NRC Inspection Manual Chapter 0609. The NRC concluded from this process that these issues have very low safety significance (Green) and no immediate safety impact. Because of the very low safety significance and because they were entered into your corrective action program, the violations are being treated as noncited violations, consistent with Section VI.A of the Enforcement Policy. If you deny the noncited violations, 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, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 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).

Sincerely,

//RA// L. E. Ellershaw for

Jeff Clark, P. E., Chief
Engineering Branch
Division of Reactor Safety

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

Enclosure:
NRC Inspection Report
50-445/04-08; 50-446/04-08

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	05000445; 05000446
Licenses:	NPF-87; NPF-89
Report No.:	05000445/2004008; 05000446/2004008
Licensee:	TXU Energy
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56 Glen Rose, Texas
Dates:	November 15 through December 3, 2004
Team Leader	J. M. Mateychick, Reactor Inspector, Engineering Branch
Inspectors:	J. P. Adams, Reactor Inspector, Engineering Branch L. E. Ellershaw, Senior Reactor Inspector, Engineering Branch B. W. Henderson, Reactor Inspector, Engineering Branch C. E. Johnson, Senior Reactor Inspector, Engineering Branch
Accompanying Personnel:	N. Morgan, Reactor Inspector, Nuclear Safety Professional Development Program J. Nadel, Reactor Inspector, Nuclear Safety Professional Development Program
Approved By:	Jeff Clark, P. E., Chief Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000445/2004-008; 05000446/2004008; 11/15-12/03/2004; Comanche Peak Steam Electric Station; Units 1 and 2; Safety System Design and Performance Capability; and Evaluation of Changes, Tests, or Experiments.

The NRC conducted an inspection with a team of five regional inspectors. The inspection identified two Green noncited violations. 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 are indicated by "No Color" or by the severity level of the applicable violation. The NRC described its program for overseeing the safe operation of commercial nuclear power reactors in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified Findings

Cornerstone: Mitigating Systems

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Section III, "Design Control," was identified for failure to maintain the design requirements for a safety class piping isolation boundary in the makeup line to the condensate storage tank. The licensee performed plant modifications and operating procedure changes, which involved a fundamental change in status of safety class piping boundary isolation valves from normally closed to normally open without determining that the new configuration did not meet the system design requirements. The issue was entered into the corrective action program as Smart Form SMF-2003-001773-00.

  The licensee had performed an operability assessment of the auxiliary feedwater system and concluded that the system remains operable, even though it is degraded because of the lack of appropriate double valve isolation between the Class III and Class V piping in the condensate storage tank makeup line. The licensee assessment showed operations personal had over 30 minutes to manually isolate a leak from the non-safety class piping. The licensee is planning to modify the condensate storage tank makeup lines to incorporate double check valve isolation meeting the appropriate design requirements for normally using the line for tank recirculation.

  The team characterized this finding as greater than minor because the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage) was affected. The finding is associated with the design control attribute of the mitigating systems cornerstone. Using the Phase 1 worksheet in Manual Chapter 0609, “Significance Determination Process”, this finding is determined to be of very low safety significance because there was no actual loss of a safety function (Section 1R21.5).

- Green. The team identified a noncited violation of 10 CFR 50.55a(f)(6)(i) for failure to fully implement NRC granted relief and alternative inservice testing requirements. Specifically, the licensee failed to perform the alternative requirement for periodic assessments, which precluded the reassessment of components to reflect changes in plant configuration, component performance test results, industry experience, and other inputs to the risk-informed process. The finding has very low safety significance and has been entered into the corrective action program as Smart Form SMF-2004--003883-00.

  The team characterized this finding as greater than minor because the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage) was affected. The finding is associated with the equipment performance attribute of the mitigating systems cornerstone. Using the Phase 1 worksheet in Manual Chapter 0609, “Significance Determination Process”, this finding is determined to be of very low safety significance because there was no actual loss of a safety function (Section 1R21.6).

B. Licensee-Identified Violations

A violation of very low safety significance was identified by the licensee and has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. The violation and corrective actions are listed in Section 40A7 of this report.

Report Details

1. REACTOR SAFETY

Introduction

The NRC performed an inspection to verify that the licensee adequately preserved the facility safety system design and performance capability and that the licensee preserved the initial design in subsequent modifications of the systems selected for review. The scope of the review also included any necessary nonsafety-related structures, systems, and components that provided functions to support safety functions. The inspection effort also reviewed the licensee's programs and methods for monitoring the capability of the selected systems to perform the current design basis functions. This inspection verified aspects of the initiating events, mitigating systems, and barrier cornerstones.

The licensee based the probabilistic risk assessment model for the Comanche Peak Steam Electric Station on the capability of the as-built safety systems to perform their intended safety functions successfully. The inspectors determined the area and scope of the inspection by reviewing the licensee's probabilistic risk analysis models to identify the most risk significant systems, structures, and components according to their ranking and potential contribution to dominant accident sequences and/or initiators. The inspectors also used a deterministic effort in the selection process by considering recent inspection history, recent problem area history, and all modifications developed and implemented.

The inspectors reviewed in detail the auxiliary feedwater system. The primary review prompted parallel review and examination of support systems, such as, electrical power, instrumentation, and related structures and components.

The inspectors assessed the adequacy of calculations, analyses, engineering processes, and engineering and operating practices that were used by the licensee to support the performance of the safety system selected for review and the necessary support systems during normal, abnormal, and accident conditions. Acceptance criteria utilized by the NRC inspection team included NRC regulations, the technical specifications, applicable sections of the Final Safety Analysis Report, applicable industry codes and standards, as well as, industry initiatives implemented by the licensee's programs.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

c. Inspection Scope

The minimum sample size for this procedure is 5 evaluations and 10 screenings. The team reviewed 7 licensee-performed 10 CFR 50.59 evaluations to verify that licensee personnel had appropriately considered the conditions under which they may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. These evaluations had been performed since the last NRC inspection of 10 CFR 50.59 activities.

The team reviewed 13 licensee-performed 10 CFR 50.59 screenings in which licensee personnel determined that evaluations were not required to ensure that exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59.

The team reviewed and evaluated a sample of six corrective action documents written since the last NRC 10 CFR 50.59 inspection to determine whether licensee personnel conducted sufficient in-depth analyses of their program to allow for the identification and subsequent resolution of problems or deficiencies.

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (71111.21)

.1 System Requirements

a. Inspection Scope

The inspectors inspected the following attributes of the auxiliary feedwater system and associated support systems: (1) process medium (water, steam, and air), (2) energy sources (ac and dc electrical systems), (3) control systems, and (4) equipment protection. The inspectors examined the procedural instructions to verify instructions were consistent with actions required to meet, prevent, and/or mitigate design basis accidents. The inspectors also considered requirements and commitments identified in the Final Safety Analysis Report, technical specifications, design basis documents, and plant drawings.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

a. Inspection Scope

The minimum sample size for this procedure is one risk-significant system for mitigating an accident. The team completed the required sample size by reviewing the auxiliary feedwater system. The primary review prompted parallel review and examination of support systems, such as, instrument air, and related structures and components.

The team assessed the adequacy of calculations, analyses, engineering processes, and engineering and operating practices that licensee personnel used for the selected safety system and the necessary support systems during normal, abnormal, and accident conditions. Acceptance criteria used by the team included NRC regulations, the technical specifications, applicable sections of the Updated Final Safety Analysis Report, applicable industry codes and standards, and industry initiatives implemented by the licensee's programs.

The team reviewed the periodic testing procedures for the auxiliary feedwater system to verify that the licensee periodically verified the capability of the system. The team also reviewed the system's operations by conducting system walkdowns; reviewing normal, abnormal, and emergency operating procedures; and reviewing the Updated Final Safety Analysis Report, technical specifications, design calculations and drawings.

The team also verified that necessary instrumentation and alarms are available to control room operators and at the remote shutdown panel, and that operators are appropriately trained in operation of the auxiliary feedwater system.

The team verified that procedures and training support local manual operation of auxiliary feedwater flow control valves should the instrument air system be lost.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed 21 smart forms written on the auxiliary feedwater system and verified that corrective actions taken were appropriately evaluated and corrected. The sample included open and closed smart forms for the past three years and are listed in the attachment to this report. Inspection Procedure 71152, "Identification and Resolution of Problems," was used as guidance to perform this part of the inspection. Older condition reports that were identified while performing other areas of the inspection were also reviewed.

b. Findings

No findings of significance were identified.

.4 System Walkdowns

a. Inspection Scope

The inspectors performed walkdowns of the accessible portions of the auxiliary feedwater system, and required support systems. The inspectors focused on the installation and configuration of switchgear, motor control centers, manual transfer switches, field cabling, raceways, piping, components, and instruments. During the walkdowns, the inspectors assessed:

- The placement of protective barriers and systems,
- The susceptibility to flooding, fire, or environmental conditions,
- The physical separation of trains and the provisions for seismic concerns,

- Accessibility and lighting for any required local operator action,
- The material condition and preservation of systems and equipment, and
- The conformance of the currently-installed system configurations to the design and licensing bases.
- The physical separation of the onsite and offsite electrical power sources.

Overall, the plant configuration was in agreement with the as-built drawings, and the external material condition of the equipment was good. The team concluded that redundancy of systems and physical separation was appropriate.

b. Findings

No findings of significance were identified.

.5 Design Review

a. Inspection Scope

The inspectors reviewed the current as-built instrument and control, electrical, and mechanical design of the auxiliary feedwater system and required support systems. These reviews included an examination of design assumptions, calculations, required system thermal-hydraulic performance, electrical power system performance, protective relaying, control logic, and instrument setpoints and uncertainties. The team specifically focused on the design basis analysis for the performance of the auxiliary feedwater pumps, such as, the design flow required, net-positive suction head, and the capacity of the condensate water storage tank to verify that a sufficient amount of water would be available during an accident. The team also reviewed the licensee's calculations and methodology for ensuring the auxiliary feedwater system was protected against seismic, flooding, fire, and high energy line break events.

The team reviewed calculations, drawings, specifications, vendor documents, Updated Final Safety Analysis Report, technical specifications, emergency operating procedures, and permanent modifications.

b. Findings

Introduction: The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Section III, "Design Control," for failure to maintain the design requirements for a safety class piping isolation boundary in the makeup line to the condensate storage tank. The licensee performed plant modifications and operating procedure changes, which involved a fundamental change in status of safety class piping boundary isolation valves from normally closed to normally open without determining that the new configuration did not meet the system design requirements. The team characterized this

violation as greater than minor because it affected the design control attribute of the Mitigating Systems Cornerstone.

Description: The original design for the condensate storage tank makeup line included a piping safety class change in the middle of the line from Class V (non-safety related) piping upstream to Class III (safety related) piping downstream attached to the condensate storage tank. The condensate storage tank makeup injection location is at the same elevation as the suction lines for the auxiliary feedwater system and if the line were to rupture and could not be isolated, there is the potential for draining the condensate storage tank to the point that the auxiliary feedwater system could not perform its safety function. To prevent this drainage, the original system design included isolation valves in the Class III portion of the line. Valves LV2478 and AF011 are parallel valves in this line upstream of Check Valve AF009. The piping safety class change occurs at the upstream side of Valves LV2478 and AF011. To meet code requirements for isolation, the licensee characterized the position of these valves as normally closed and characterized, by analysis, Valve AF009 as exempt from single failure based on its being normally closed. This was consistent with the original design requirements, as Valve AF011 is a manual valve that was maintained normally closed, Valve LV2478 is an air-operated flow control valve that is only opened during condensate storage tank makeup operations (an infrequent operation), and Valve AF009 is a check valve that is only opened when there is flow in the line (i.e., only during condensate storage tank makeup operations.)

The licensee changed the status of Valves AF009 and AF011 from normally closed to normally open, violating their design basis for isolation of the condensate storage tank safety related piping from non-safety related piping. The licensee did this in an attempt to control the oxygen concentration in the condensate storage tank, first by treating the water in ion exchange columns and, finally, by continuously recirculating a small flow of water from the condensate storage tank. The licensee selected the non-safety related part of the condensate storage tank makeup line as the injection point of the recirculating flow. Prior to this change, flow in this line only occurred when the licensee made up mass to the condensate storage tank, an occasional evolution. Since making this change, the licensee maintains flow in this line on a nearly continuous basis and these valves are, therefore, now normally open.

Analysis: The licensee modified the condensate storage tank makeup line, including changing the operation of valves in the safety related piping, without consideration of the effect this would have on the safety function of Valves AF009 and AF011. The licensee made this modification over a period of several years and included several system hardware and procedure changes. At each stage in the process, the licensee failed to consider the effect of this change on the status of Valves AF009 and AF011, including the degradation of their safety function.

The team considered this to be a performance deficiency because the licensee should have, but failed to maintain the system design requirements during the process. The team determined that this violation is more than minor because it affected the design control attribute of the Mitigating Systems Cornerstone. Failure to maintain the design requirements for a safety related piping isolation boundary in the makeup line to the

condensate storage tank affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The team evaluated the significance of the violation using the significance determination process. The team performed a Phase I significance determination process evaluation and concluded that the violation is of very low safety significance (i.e., significance determination process Green), based on the first question under Mitigating Systems in the Phase 1 worksheet of Inspection Manual Chapter 0609, Appendix A (i.e., Is the finding a design or qualification deficiency that has been confirmed to result in a loss-of-function in accordance with Generic Letter 91-18 (Revision 1)?).

The licensee had performed an operability assessment of the auxiliary feedwater system and concluded that the system remains operable, even though it is degraded because of the lack of appropriate double valve isolation between the Class III and Class V piping in the condensate storage tank makeup line. Since the licensee has entered the degraded condition into their corrective action program (Smart Form SMF-2003-001773-00) and is scheduled to modify the condensate storage tank makeup lines to incorporate double check valve isolation, the team has characterized this violation as a noncited violation.

Enforcement: Final Safety Analysis Report, Section 9.2.6, “Condensate Storage Facilities,” Subsection 9.2.6.2, “System Description,” states, “Inadvertent drainage of the tank is prevented by normally closed double valves.” Additionally, the licensee designated Check Valve AF009 as exempt from the single failure criterion on the basis that it is a normally closed check valve, in accordance with the requirements in Procedure DBD-ME-028 “Design Basis Document Classification of Structures, Systems and Components,” Section 5.5(d).

Appendix B, Section III of 10 CFR Part 50 states, “Measures shall be established to assure that applicable regulatory requirements and the design basis . . . are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. . . . Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces. . . . Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design . . . .”

Contrary to the above, the licensee inadvertently changed the status of Valves AF009 and AF011 from normally closed to normally open without considering the safety function of these valves. The revised configuration did not meet the original design requirements of the system, and appropriate system modifications to maintain required isolation of safety class piping from non-nuclear safety grade piping were not identified, thus, demonstrating a lack of control of their design basis. Based on the very low safety significance of this violation and because the finding was entered into the licensee’s corrective action program, the team treated this as a noncited violation, consistent with

Section VI.A of the NRC Enforcement Policy: NCV (05000445; 446/2004008-001), Failure to maintain design control over a safety related boundary isolation.

.6 Safety System Inspection and Testing

a. Inspection Scope

The team reviewed the program and procedures for testing and inspecting selected components for the auxiliary feedwater system and required support systems. The review included the results of surveillance tests required by the technical specifications and selective review of inservice tests.

b. Findings

Introduction: The team identified a Green noncited violation of 10 CFR 50.55a, "Codes and Standards," for the licensee's failure to fully comply with an NRC-approved request for relief, which authorized use of a risk-informed inservice test program alternative to Section XI of the ASME Code.

Description: Relief Request A-1, submitted by licensee personnel, proposed an alternative to 10 CFR 50.55a(f)(4)(ii). The proposed alternative was a risk-informed process to determine the safety significance and testing strategy of components in the ASME Section XI, "IST Program," and identify non-ASME inservice test components (pumps and valves) modeled in probabilistic risk analysis that are determined to be high safety significant components. The risk-informed process consisted of nine numerically identified elements, one of which was periodic re-assessment.

Element 8 referred to periodic assessments, and stated that components will be re-assessed at a frequency not to exceed every other refueling outage (based on Unit 1 refueling outages) to reflect changes in plant configuration, component performance test results, industry experience, and other inputs to the process. This re-assessment will be completed within 9 months of completion of the outage.

It further stated that part of this periodic re-assessment would be a feedback loop of information to the probabilistic risk analysis. This would include information, such as, components tested since last re-assessment, number and types of tests, number of failures, corrective actions taken, including generic implication and changed test frequencies. Once the probabilistic risk analysis has been re-assessed, the information would be brought back to the integrated decision process (the function of which is to ensure the risk ranking input information is consistent with plant design, operating procedures, and with plant-specific operating experience). The integrated decision process is to review and confirm the existing lists of high safety significant components and low safety significant components or modifications of these lists based on the new data. Confirmatory measures previously utilized to categorize components as low safety significant components will be validated and the maximum test intervals will be verified or modified as dictated by the integrated decision process. The relief request was reviewed and accepted by the NRC in a safety evaluation report dated August 14, 1998.

The team requested the periodic assessments performed to date. The licensee informed the team that the required periodic assessments of the risk-informed inservice test program had not been performed, and initiated Smart Form SMF-2004-003883-00 to address this issue and document the proposed corrective actions.

Analysis: The team identified this as a performance deficiency for the failure to implement the alternative requirements to 10 CFR 50.55a(f)(4)(ii) as allowed by paragraphs (f)(5)(iii) and (f)(6)(i) of 10 CFR 50.55a.

The team determined, through Inspection Manual Chapter 0612, Appendix B, that this finding is more than minor, in that, it affected the equipment performance attribute of the Mitigating Systems Cornerstone. The failure to implement the alternative requirements to 10 CFR 50.55a(f)(4)(ii) precluded the re-assessment of components to reflect changes in plant configuration, component performance test results, industry experience, and other inputs to the risk-informed process and affected the Mitigating Systems Cornerstone objective to assure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The team evaluated the significance of the violation using the significance determination process. The team performed a Phase I significance determination process evaluation and concluded that the violation is of very low safety significance (i.e. significance determination process Green), based on the first question under Mitigating Systems in the Phase 1 worksheet of Inspection Manual Chapter 0609, Appendix A (i.e., Is the finding a design or qualification deficiency that has been confirmed to result in a loss-of-function in accordance with Generic Letter 91-18 (Revision 1)?).

Enforcement: Paragraph (f)(4)(ii) in 10 CFR 50.55a states, in part, “Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety . . . must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section. . . .” Paragraph (f)(5)(iii) of 10 CFR 50.55a states, in part, “If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit . . . information to support the determination.” Paragraph (f)(6)(i) of 10 CFR 50.55a states, in part, “The Commission will evaluate the determinations under paragraph (f)(5) of this section that code requirements are impractical. The Commission may grant relief and may impose alternative requirements as it determines is authorized by law . . . .”

Contrary to the above, the licensee failed to fully implement the Commission granted relief and alternative requirements. Because the finding is of very low safety significance and has been entered into the licensee’s corrective action program as Smart Form SMF-2004-003883-00, the team treated this as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV (05000445; 446/2004008-002), Failure to fully implement Commission granted relief and alternative requirements.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of problems associated with the Auxiliary Feedwater System and required support systems that were identified by licensee personnel in the corrective action program to evaluate the effectiveness of corrective actions related to design issues. The sample included open and closed condition reports for the past 3 years, which are listed in the attachment to this report. Inspection Procedure 71152, “Identification and Resolution of Problems,” was used as guidance to perform this part of the inspection. Older condition reports that were identified while performing other areas of the inspection were also reviewed.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

The team leader presented the inspection results to Mr. R. Flores, Vice President Nuclear Operations, and other members of licensee management at the conclusion of the onsite inspection on December 3, 2004.

At the conclusion of this meeting, the team leader asked the licensee's management whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

40A7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an noncited violation.

Part 50.59(c)(1) of Title 10 states, “A licensee may make changes in the facility as described in the final safety analysis report (as updated), . . . without obtaining a license amendment pursuant to § 50.90 only if: . . . (ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section. . . .(d)(1) The licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation, which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2) of this section."

Contrary to the above, the licensee modified the Condensate storage tank makeup line in a manner that did not meet the design requirements as described in the Final Safety Analysis Report without performing a valid 50.59 evaluation of the change. Specifically, the original design for the condensate storage tank makeup line included a piping safety class change in the middle of the line from Class V (non-safety related) piping upstream to Class III (safety related) piping downstream attached to the condensate storage tank. In the effort to establish an effective oxygen control process for the condensate storage tank (which is the source of water for the auxiliary feedwater system), the licensee changed the status of the line from normally isolated to normally open having a tank recirculation flow through it. In the new operating configuration, the condensate storage tank makeup line valves were no longer in the positions required for isolation of the non-nuclear safety class piping.

As the oxygen control process for the condensate storage tank evolved, the licensee performed multiple system modifications over a period of years. Although the licensee performed a 50.59 screening at each stage in the evolution of the condensate storage tank oxygen control process, they incorrectly concluded that a 50.59 evaluation was not required. Therefore, they did not perform a valid 50.59 evaluation for the change. The licensee identified the degraded condition during a walkdown of the auxiliary feedwater system/condensate storage tank system in 2003. The licensee performed an operability assessment and concluded that this degraded condition did not result in the auxiliary feedwater system being inoperable. This finding is of very low safety significance and was documented in the licensee's corrective action program as Smart Form SMF-2003-001773-00.

ENCLOSURE

KEY POINTS OF CONTACT

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NRC

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

ITEMS OPENED AND CLOSED

Opened and Closed

05000445,446/2004008-001	NCV	Failure to maintain design control over a safety class boundary isolation
05000445,446/2004008-002	NCV	Failure to fully implement Commission granted relief and alternative requirements

LIST OF DOCUMENTS REVIEWED

The following documents were selected and reviewed by the team to accomplish the objectives and scope of the inspection.

CALCULATIONS

Number	Title	Revision
16345-IC(B)-002	Air Accumulator Sizing	4
2-IC-0001	Air Accumulator Sizing	0 & CCN 001
2-ME-0054	Auxiliary Feedwater Pumps NPSH and Minimum Acceptable Service Water Supply Pressure to Auxiliary Feedwater	0 & CCN 002
2-ME-0063	Condensate Storage Tank TS Limits	0
2-ME-0076	Auxiliary Feedwater System Performance	0
16345-ME(B)-143	Maximum DP For Which Auxiliary Feedwater System MOV's Must Be Operable	2
ICB008	Calculation Change Notice	3
ICB009	Calculation Change Notice	3
ICB059	Calculation Change Notice	3
ICB060	Calculation Change Notice	3
ICS607	Calculation Change Notice	0
ICS608	Calculation Change Notice	0
ME-(B)-186	Verification of TDAFW Pump Performance At Low End Steam Conditions	1
ME(B) 240	Condensate Storage Tank TS Limits	2
ME (B) 241	AFW Pumps Technical Specification Limits	2
ME (B) 349	Main Steam Supply to the Turbine AFW Pump	0, 1
ME-CA-0000-1093 Attachment J, page 1-HV-2491A*1	MOV Tag - 1-HV-2491A Setup and Design Margin Calculation	7

Number	Title	Revision
ME-CA-0000-1093 Attachment J, page 1-HV-2480*1	MOV TAG: 1-HV-2480 Setup and Design Margin Calculation	7
ME-CA-0000-1093 Attachment J, page 1-HV-2481*1	MOV TAG: 1-HV-2481 Setup and Design Margin Calculation	7
ME-CA-0000-1093 Attachment J, page 1-HV-2482*1	MOV TAG: 1-HV-2482 Setup and Design Margin Calculation	7
ME-CA-0000-3342	Air Accumulator Check Valve Leakage - Decay Rate, Pressure and Time	1
ME-CA-0206-5085	Evaluation of the Effects of Increasing the Motor Driven Auxiliary Feedwater Minimum Recirculation Flow to 200 gpm	0
RXE-TA-CPX/0-053	TDAFW Turbine System Redesign Analysis	3

DESIGN BASIS DOCUMENTS

Number	Title	Revision
DBD-ME-028	Classification of Structures, Systems and Components	11
DBD-ME-202	Main Steam, Reheat and Steam Dump System	14
DBD-ME-206	Auxiliary Feedwater System	17

DRAWINGS

Number	Title	Revision
M1-0206	Flow Diagram, Auxiliary Feedwater System	CP-19
M1-0206, Sheet 01	Flow Diagram, Auxiliary Feedwater System Pump Trains	CP-13
M1-0206, Sheet 01	Flow Diagram, Auxiliary Feedwater System Yard Layout	CP16
M1-0218, Sheet 01A	Flow Diagram Instrument Air Safeguards Building	CP-19
M2-0202	Flow Diagram Main Steam Reheat And Steam Dump	CP-22
M2-0202	Flow Diagram Main Steam Reheat And Steam Dump	CP-2/03

Number	Title	Revision
M2-0206	Flow Diagram Auxiliary Feedwater Pump Trains	CP-9/01
M2-0206	Flow Diagram Auxiliary Feedwater System Yard Layout	CP-6
M2-0206	Flow Diagram Auxiliary Feedwater System	CP-14
M2-0206	Flow Diagram Auxiliary Feedwater System Yard Layout	CP-16

10 CFR 50.59 SCREENINGS

Number	Revision	Number	Revision
59SC-1999-001773-01-04	4	59SC-2002-003296-01-00	0
59SC-1999-002473-03-00	0	59SC-2002-003920-01-00	0
59SC-2000-001669-01-01	1	59SC-2002-004167-01-00	0
59SC-2000-002171-01-00	0	59SC-2002-004167-02-00	0
59SC-2000-003314-01-00	0	59SC-2002-004218-01-00	0
59SC-2002-000447-02-00	0	59SC-2003-000262-01-00	0
59SC-2002-002473-02-00	0		

10 CFR 50.59 Evaluations

Number	Revision	Number	Revision
SE 99-33	0	59EV-2002-000100-01-00	0
50EV-2000-002255-01-00	0	59EV-2002-000129-01-00	0
59EV-2001-000158-01-00	0	59EV-2002-001062-01-00	0
59EV-2001-001672-01-00	0	59EV-2003-002742-01-00	0

MODIFICATIONS

Number	Title	Revision
DCN 10888	Relocation of Valve 2AF-0326	0
DCA 94129	Revise Valve Symbols	

PROBLEM IDENTIFICATION REPORTS (Smart Forms)

SMF-1999-000118-00	SMF-2002-002925-00	SMF-2004-003120-00
SMF-2000-000021-00	SMF-2002-003690-00	SMF-2004-003443-00
SMF-2000-001232-00	SMF-2003-000556-00	SMF-2004-003444-00
SMF-2001-000643-00	SMF-2003-000635-00	SMF-2004-003495-00
SMF-2002-000560-00	SMF-2003-001068-00	SMF-2004-003538-00
SMF-2002-001754-00	SMF-2003-001379-00	SMF-2004-003769-00*
SMF-2002-001760-00	SMF-2003-001454-00	SMF-2004-003754-00*
SMF-2002-001814-00	SMF-2003-001773-00	SMF-2004-003775-00*
SMF-2002-002222-00	SMF-2003-002314-00	SMF-2004-003782-00*
SMF-2002-002234-00	SMF-2003-003479-00	SMF-2004-003787-00*
SMF-2002-002244-00	SMF-2004-000085-00	SMF-2004-003790-00*
SMF-2004-002294-00	SMF-2004-000525-00	SMF-2004-003854-00*
SMF-2002-002773-00	SMF-2004-002605-00	SMF-2004-003883-00*
SMF-2002-002851-00	SMF-2004-002698-00	

*Initiated as a result of inspection activities.

PROCEDURES

Number	Title	Revision
ABN-301	Instrument Air System Malfunction	10
ABN-305	Auxiliary Feed System Malfunction	5
CLI-814	Operation of Recirculation Skid (CST)	2
CLI-814A	Operation of Deox Skid (CST)	0
CLI-814A	Operation of Deox Skid (CST)	1
ECA-0.0B	Loss of All AC Power	1
ECA-2.1B	Uncontrolled Depressurization of All Steam Generators	1

Number	Title	Revision
ECE-5.01	Design Control Program	7
ECE-5.03	Calculations	15
ECE-5.01-08	Electronic Design Change Process	6
EOP-0.0B	Reactor Trip of Safety Injection	1
EOP-0.1B	Loss of Reactor or Secondary Coolant	1
EOP-2.0B	Faulted Steam Generator Isolation	1
EOP-3.0B	Steam Generator Tube Rupture	1
EOS-1.1B	Safety Injection Termination	1
FRH-0.1B	Response To Loss of Secondary Heat Sink	1
INC-2010	Valve Calibration Using Fisher Flowscanner Control Valve Diagnostic System	1
INC-2012	Valve Calibration Fisher Controls Type 657 Air-to-Close Valve Actuators	4
INC-4416A	Channel Calibration Motor Driven Auxiliary Feedwater Pump 01 Discharge and Recirculation Flow Channel 2456	3
OPT-206A	8.3.3 TD AFW PMP 1-01 Operability Test	24
OPT-206B	AFW SYSTEM	16
OPT-224A	Reactor Makeup Water System	9
OPT-430A-1	Train A Safety Injection With Loss Of Offsite Power	2
OPT-435A-1	Train B Safety Injection With Loss Of Offsite Power	2
OPT-502A	Auxiliary Feed Water/Station Service Water Section XI Cross-Tie Valves	7
OPT-506A	CPSES Unit 1 Operating Testing Manual, Quality Related AFW System	5
OPT-506B	CPSES Unit 2 Operating Testing Manual, Quality Related AFW System	8
OPT-530A	Auxiliary Feedwater Check Valve Reverse Flow Test	0
OPT-601A	Train A MDAFW Accumulator Check Valve Leak Test	3
OPT-601B	Train A MDAFW Accumulator Check Valve Leak Test	3

Number	Title	Revision
OPT-602A	Train B MDAFW Accumulator Check Valve Leak Test	3
OPT-602B	Train B MDAFW Accumulator Check Valve Leak Test	3
OPT-603A	TDAFW Accumulator Check Valve Leak Test	4
OPT-603B	TDAFW Accumulator Check Valve Leak Test	3
PPT-P0-6004	Safety-Related Rising Stem MOV Testing MOV Tag 1-HV-2491A-C	1
PPT-P0-6004	Safety-Related Rising Stem MOV Testing MOV Tag 1-HV-2480	1
PPT-S0-6004	Motor Operated Rising Stem Valve Risk-Informed IST Testing MOV Tag 1-HV-2481	1
PPT-S0-6004	Motor Operated Rising Stem Valve Risk-Informed IST Testing MOV Tag 1-HV-2482	1
PPT-S2-9104B	TDAAFAW Pump Actuation And Response Time Test, Train B	2
SOP-304A	Auxiliary Feedwater System	15
STA-202	Administrative control of CPSES Nuclear Generation Procedures	29
STA-205-R20-1	Quality Related Changes to Procedures	20
STA-421	Initiation of Smart Forms	10
STA-422	Processing Smart Forms	19
STA-504	Technical Evaluation	10 & PCF 2
STA-707	10 CFR 50.59 Reviews	16

TECHNICAL EVALUATIONS

Number	Title	Revision
TE-95-226	Technical Evaluation - Acceptance Criteria for Accumulator Check Valve Leakage Criteria	0

TRAINING MANUALS

Number	Title	Revision
OP51.SYS.AF1	Training Material - Auxiliary Feedwater System	8-23-04
OP51.SYS.AF1	Training Material - Auxiliary Feedwater System	Change 1

MISCELLANEOUS DOCUMENTS

Number	Title	Revision
ASME Code Case-504-2	Alternative Rules for Repair of Classes 1, 2, and 3 Austenitic Stainless Steel Piping	March 12, 1997
CPSES-200203940	Comanche Peak Steam Electric Station (CPSES) Unit 1 Docket No. 50-445 Relief Request B -3 for Second 10 Year ISI Interval from 10 CFR 50.55a Requirements for Class 1 The Repair/Replacement of CRDM Canopy Seal Weld (Interval Start Date: August 13, 2000, Second Interval)	December 6, 2002
CPSES-200300265	Comanche Peak Steam Electric Station (CPSES) Unit 2 Docket No. 50-446 Relief Request B -10 for Second 10 Year ISI Interval from 10 CFR 50.55a Requirements for Class 1 The Repair/Replacement of CRDM Canopy Seal Weld (Interval Start Date: August, 2003, Second Interval)	February 11, 2003
FSAR Section 10.3.2.6	Auxiliary Feedwater Pump Turbine Steam Supply	Amendment 97
FSAR Section 10.4.9	Auxiliary Feedwater System	Amendment 98
Letter from Bartholomew C.Buckley (NRC) to George A. Hunger (PECO)	Limerick Generating Station (LGS), Units 1 and 2 of Main turbine Rotor Replacement, Extension of Turbine Rotor Inspection Intervals and Valve Testing Frequencies (TAC Nos. M99341 and M99342)	February 3, 1998

Letter from Charles E. Rossi (NRC) to James A. Martin (WEC)	Approval for Referencing of Licensing Topical Reports WSTG-1-P, May 1981, 'Procedures for Estimating the Probability of Steam Turbine Disc Rupture from Stress Corrosion Cracking,', March 1974, 'Analysis of the Probability of the Generation and Strike of Missiles from a Nuclear Turbine', WSTG-2-P, May 1981, 'Missile Energy Analysis Methods for Nuclear Steam Turbines', and WSTG-3-P, July 1984, 'Analysis of the Probability of a Nuclear Turbine Reaching Destructive Overspeed'"	February 2, 1987
Letter from Herbert N. Berkow (NRC) to Stan Dembkowski (SWPC)	Safety Evaluation for Acceptance of Referencing the Siemens Westinghouse Topical Report, 'Missile Analysis Methodology for General Electric (GE) Nuclear Steam Turbine Rotors by the Siemens Westinghouse Power Corporation (SWPC)' TAC No. MB5679.	July 22, 2003
LDCR TB-99-7	Revises SR 3.7.5.2 AFW required differential pressures	
LDCR SA-97-53	Unit 1 AFW Steam Admission Valve Protective Train Swap DM 97-15	
	CPSES 50.59 Resource Manual	2
	Safety Evaluation by the Office of Nuclear Reactor Regulation, The Request for Relief No. B-3 Second 10-year Interval Inservice Inspection Program Plan, Comanche Peak Steam Electric Station Unit 1, TXU Generation Company LP, Docket No. 50-445	January 3, 2003
	Safety Evaluation by the Office of Nuclear Reactor Regulation, Second 10-year Interval Inservice Inspection Interval Request for Relief, TXU Generation Company LP, Comanche Peak Steam Electric Station Unit 2, Docket No. 50-446	March 21, 2003
Standing Order OSO-003	Long Term Evaluated Work-Arounds	0

Work Orders

5-98-504298-AA	5-01-505436-AA	5-03-504171-AA, -AB
5-99-500175-AA	5-01-505439-AA	5-03-505435-AA
5-99-501147-AF	5-01-505440-AA	5-03-504440-AG, -AI, -AK
5-99-504460-AA	5-01-505443-AA	5-03-505445-AA

5-00-500977-AC	5-01-505518-AA	5-03-505611-AA
5-00-501147-AC	5-01-505519-AA	5-03-505822-AA
5-00-504469-AC	5-01-505825-AA	5-03-505824-AA
5-00-504491-AC	5-02-501037-AA	5-04-504438-AJ
5-00-504556-AD	5-02-505379-AA	5-04-504438-AE, -AG, -AH, -AI, -AJ
5-00-504635-AA	5-02-505541-AA	5-04-504441-AD, -AE, -AF, -AG, -AH, -AI, -AJ
5-01-505313-AA	5-02-505563-AA	5-04-505583-AE, -AF
5-01-505385-AA	5-02-505608-AA	5-04-505592-AC
5-01-505434-AA	5-02-505610-AA	5-04-505610-AC, -AD