TUELECTRIC

Log # TXX-91415
File # 10200
Ref. # 50.73(a)(2)(vii)

November 29, 1991

William J. Cahill, Jr.
Group Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
INOPERABLE TRAIN/CHANNEL IN SAFETY RELATED SYSTEM
LICENSEE EVENT REPORT 91-026-00

Gentlemen:

Enclosed is Licensee Event Report 91-026-00 for Comanche Peak Steam Electric Station Unit 1, "Pressurizer Safety Valves Discovered to be Inoperable using New Test Methodology."

Sincerely,

William J. Cahill, Jr.

By: [signature]
W. G. Guldemond
Manager, Site Licensing

OB/tg

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (2)

JE22 1/1

Enclosure to TXX-91415

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

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Title (4)
PRESSURIZER SAFETY VALVES DISCOVERED TO BE INOPERABLE USING NEW TEST METHODOLOGY

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Numbers
									N/A	0 5 0 0 0
1 0	3 0	9 1	9 1	- 0 2 6 -	0 0	1 1	2 9	9 1	N/A	0 5 0 0 0

Operating Mode (9): 6
Power Level (10): 0 0 0

This report is submitted pursuant to the requirements of 10 CFR §: (Check one or more of the following) (11)

☐ 20.402(b)	☐ 20.405(c)	☐ 50.73(a)(2)(iv)	☐ 73.71(b)
☐ 20.405(a)(1)(i)	☐ 50.36(c)(1)	☐ 50.73(a)(2)(v)	☐ 73.71(c)
☐ 20.405(a)(1)(ii)	☐ 50.36(c)(2)	☒ 50.73(a)(2)(vii)	☐ Other (Specify in Abstract below and in Text, NRC Form 366A)
☐ 20.405(a)(1)(iii)	☐ 50.73(a)(2)(i)	☐ 50.73(a)(2)(viii)(A)	
☐ 20.405(a)(1)(iv)	☐ 50.73(a)(2)(ii)	☐ 50.73(a)(2)(viii)(B)	
☐ 20.405(a)(1)(v)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(x)	

Licensee Contact For This LER (12)

Name	Telephone Number (Area Code)
D.E. BUSCHBAUM COMPLIANCE SUPERVISOR	8 1 7 8 9 7 - 5 8 5 1

Complete One Line For Each Component Failure Described in This Report (13)

Cause	System	Component	Manufacturer	Reportable To NPRDS	Cause	System	Component	Manufacturer	Reportable To NPRDS

Supplemental Report Expected (14)

☐ Yes (If yes, complete Expected Submission Date) ☒ No

Expected Submission Date (15): Month | Day | Year

Abstract (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On October 18, 1988, all three Pressurizer Safety Valves (PSV) were tested and adjusted at the Crosby Valve and Gage Company using 200 ± 10 degree Fahrenheit water at the valve inlet, approximating installed plant conditions for PSVs with hot water loop seals. On October 28, 1991, using the state-of-the-art test methodology (saturated steam as the test medium), all three PSV were found to have setpoints one to four percent less than the setpoint tolerance allowed by Technical Specifications (TS). All three PSV setpoints were subsequently reset to TS limits.

The root cause of this event was a less than adequate PSV test methodology/procedure. Corrective action included testing of the PSVs to state-of-the-art methodology and resetting the PSVs to TS limits.

Enclosure to TXX-91415

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple safety related systems or two independent trains or channels to become inoperable in a single safety related system.

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On October 30, 1991, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 6, Refueling, with the Reactor Coolant System (RCS)(EIIS:(AB)) at a temperature of 100 degrees Fahrenheit and approximately atmospheric pressure.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems or components that contributed directly to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

On October 18, 1988, all three CPSES Unit 1 Pressurizer Safety Valves (PSV)(EIIS:(RV)(AB)) were tested and adjusted at the Crosby Valve and Gage Company. During this test the PSVs were adjusted to Technical Specification (TS) setpoint requirements using 200 ± 10 degree Fahrenheit water at the valve inlet, approximating installed plant conditions for valves with hot water loop seals. The test methodology used was consistent with the CPSES commitment to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME), Section XI, 1980 Edition through Winter 1981 Addenda. In December, 1988, CPSES committed to ASME Section XI, 1986 Edition. This new edition references American National Standards Institute (ANSI) /ASME OM-1-1981 for test frequency, preservice, periodic, and test method requirements.

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Text (If more space is required, use additional NRC Form 366A's) (17)

In July and October, 1989, industry problems with PSV setpoint shift were documented by Westinghouse. On December 28, 1989, the Nuclear Regulatory Commission issued Information Notice IN 89-90, "Pressurizer Safety Valve Lift Setpoint Shift." On November 6, 1990, CPSES loaned a spare PSV to the Westinghouse Owners Group (WOG) to be used in the testing program to characterize PSV setpoint shift. Findings of the WOG test program were used to develop a testing program for the CPSES Unit 1 PSVs. As a result, a new test methodology was developed using saturated steam at the valve inlet.

On October 23, 1991, all three PSVs were sent to the Westinghouse Western Service Center for setpoint testing. The CPSES PSV testing program tested all three PSVs, even though only one PSV was required to be tested to satisfy TS surveillance and inservice testing requirements. On October 28 and 29, 1991, testing was performed using a procedure designed to comply with ANSI/ASME OM-1-1981 and WOG guidance, using saturated steam as the test medium. As a result of the test, all three PSVs were found to have setpoints one to four percent less than the setpoint tolerance allowed by TS 3.4.2.1. All three PSV setpoints were subsequently reset to TS limits. On October 30, 1991, the CPSES Inservice Test (IST) Coordinator (non-licensed, contractor) was reviewing the PSV test data when all three PSVs were discovered to have been in violation of TS requirements. The IST Coordinator promptly documented this discovery in accordance with plant procedures.

E. <u>THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE, OR PROCEDURAL OR PERSONNEL ERROR</u>

The PSVs were being tested to satisfy the requirements of the CPSES Unit 1 PSV testing program and to satisfy Technical Specification surveillance and inservice testing requirements. The unsatisfactory lift setpoints were discovered as the result of this test.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

Not applicable - there were no component failures associated with this event.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Not applicable - there were no component failures associated with this event.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF COMPONENTS WITH MULTIPLE FUNCTIONS

Not applicable - there were no failed components with multiple functions that affected this event.

D. FAILED COMPONENT INFORMATION

Not applicable - there were no component failures associated with this event.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

Not applicable - no safety system responses occurred as a result of this event.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

The three PSVs were set to TS limits using the 1988, test methodology, and were considered operable. In 1991, using the new test methodology, the PSVs were determined to be inoperable. The PSVs were originally set to TS limits on October 18, 1988, and found to be inoperable on October 30, 1991 (3 years, 12 days). Although the PSV lift setpoints were less than the setpoint tolerance allowed by TS, the PSVs were still capable of performing their intended safety function.

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C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The PSVs operate to prevent the RCS from being pressurized above its Safety Limit of 2735 pounds per square inch-gage. Each safety valve is designed to relieve 420,000 pounds per hour of saturated steam at the valve setpoint. The combined relief capacity of all three PSVs is greater than the maximum surge rate resulting from a complete loss-of-load assuming no reactor trip until the first Reactor Trip System (EIIS:(JC)) setpoint is reached and also assuming no operation of the Pressurizer Power Operated Relief Valves (PORV)(EIIS:(RV)(AB)) or Steam Dump Valves (EIIS:(V)(SB)).

The reduction of the PSV lift setpoint does not adversely affect the licensing basis analysis, provided the lift setpoint remains above the PORV lift setpoint. The basis for this is that each transient is evaluated to assess whether operation of the PORV results in a more limiting transient relative to the event acceptance criteria. Thus, provided that the PSV opens at a pressure greater than the PORV set pressure, the early opening of the PSV would result in a less limiting case.

The difference between the PORV setpoint and the PSV setpoint is approximately six percent; hence the reduction in the PSV lift setpoint did not adversely affect the conclusions of the accident analyses. Furthermore; the relief capacity of the PSVs at nominal setpoint was not reduced by the reduction in the lift setpoint.

IV. CAUSE OF THE EVENT

ROOT CAUSE

The root cause of this event was a less than adequate PSV test methodology/procedure. In 1988, the PSV setpoint shift problems were not known. All previous recommendations indicated that the PSVs should be tested by simulating installed conditions. As industry problems arose, and the PSV setpoint shift issue was being evaluated, new test methods were developed. Using state-of-the-art test methods, the PSV setpoints were discovered to be less than the setpoint tolerance allowed by TS 3.4.2.1.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

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Text (If more space is required, use additional NRC Form 366A's) (17)

V. CORRECTIVE ACTIONS

A. CORRECTIVE ACTIONS TO PREVENT RECURRENCE

ROOT CAUSE

Less than adequate PSV test methodology/procedure.

CORRECTIVE ACTION

New PSV test methodology has been implemented. All three PSVs were tested to state-of-the-art requirements identified through the WOG program. The PSV setpoints have been reset to TS limits.

VI. PREVIOUS SIMILAR EVENTS

There have been no previous similar events reported pursuant to 10CFR50.73.

VII. ADDITIONAL INFORMATION

The times listed in the report are approximate and Central Standard Time.