TUELECTRIC

Log # TXX-94303
File # 10200
Ref. # 50.73(a)(2)(vii)

C. Lance Terry
Group Vice President

November 21, 1994

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) - UNIT 2
DOCKET NOS. 50-446
INOPERABLE TRAIN/CHANNEL IN SAFETY RELATED SYSTEM
LICENSEE EVENT REPORT 446/94-018-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 94-018-00 for Comanche Peak Steam Electric Station Unit 2, "Pressurizer Safety Valves Discovered to be Inoperable From As-Found Testing."

Sincerely,

C. L. Terry

AQ:tg
Enclosure

c - Mr. L. J. Callan, Region IV
Mr. D. D. Chamberlain, Region IV
Resident Inspectors, CPSES

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Enclosure to TXX-94303

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION	APPROVED OMB NO.3150-0104 EXPIRES: 4/30/92
LICENSEE EVENT REPORT (LER)	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.

Facility Name (1)	Docket Number (2)	Page (3)
COMANCHE PEAK-UNIT 2	0 5 0 0 0 4 4 6	1 OF 5

Title (4)

PRESSURIZER SAFETY VALVES DISCOVERED TO BE INOPERABLE FROM AS-FOUND TESTING

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Numbers
1 0	2 5	9 4	9 4	- 0 1 8	- 0 0	1 1	2 1	9 4	N/A	0 5 0 0 0
									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	Area Code	Telephone Number
W.G. Guldemond, Systems Engineering Manager	8 1 7	- 8 9 7 - 8 7 3 9

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)		Expected Submission Date (15)	Month	Day	Year
☐ Yes (if yes, complete Expected Submission Date)	☒ No				

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

On October 25, 1994, using saturated steam as the test medium, three Pressurizer Safety Valves (PSV) were found to have setpoints outside the setpoint tolerance allowed by Technical Specifications (TS). All three PSVs were subsequently reset within TS limits.

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 25, 1994, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 6, Refueling, with the Reactor Coolant System (RCS)(EIIS:(AB)) at a temperature of 87 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

In October of 1994, all three Pressurizer Safety Valves (PSVs) had been returned to the Westinghouse Western Service Center for surveillance testing in support of the first refueling outage for CPSES Unit 2. Testing was performed using a procedure designed to comply with ANSI/ASME OM-1987 Part 1 and WOG guidance, using saturated steam as the test medium. On October 25, 1994, PSV 2-8010C was found to be 2.012% above the Technical Specification 3.4.2.2 setpoint which specifies a plus or minus 1% range. As a result, the remaining PSVs were tested on October 26, 1994. One failed at 1.046% above the setpoint, the other failed at 1.126% above the setpoint. All three PSVs were subsequently reworked and the relief setpoints were verified to be within TS limits.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE, OR PROCEDURAL OR PERSONNEL ERROR

The PSVs were being tested to satisfy the requirements of the CPSES Inservice Testing Plan and to satisfy Technical Specification (TS) surveillance 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 initially set to within TS limits on November 11 and 12, 1992, and were considered operable until they were determined to be inoperable on October 25 and 26, 1994. Although the PSV lift setpoints were out of the TS range, the PSVs were still capable of fulfilling their safety function.

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 Relief Valves (PORV)(EIIS:(RV)(AB)) or Steam Dump Valves (EIIS:(V)(SB)).

In the accident analyses, each of the three PSVs is assumed to open at their nominal set pressure, based on the expectation of a random distribution of as-found set pressures about the nominal set pressure. The CPSES-2 as-found PSV set pressures satisfy this expectation. Further, an allowance for accumulation of +3% of the nominal set pressure is provided in the accident analyses; i.e., the PSVs are not assumed to be fully open until the pressure increases to 103% of the nominal set pressure. PSV 2-8010C was fully open at 102% of the nominal set pressure, PSV 2-8010A was fully open at 101% of the nominal set pressure and PSV 2-8010B was fully open at 101% of the nominal set pressure. Hence, the assumptions of the accident analysis remain valid. Also, the actual relief capacity of the PSVs was not affected and the PSVs could have fulfilled their safety function with the PSVs slightly deviated from their required setpoint range. The lift setpoints remained above the PORV lift setpoints. In conclusion, these Unit 2 PSVs did not meet setpoint criteria required by CPSES Technical Specifications by a narrow margin. No challenges to operate the PSVs were presented by the RCS. The functional capacity of these PSVs was not affected and the health and safety of the public was unaffected.

IV. CAUSE OF THE EVENT

A specified cause of the observed setpoint drift could not be determined. The valve manufacturer has indicated in the past that the setpoint changes do not indicate a material problem with the valves. The test results were within the 3 percent acceptance range of ASME/ANSI OM-1987 Part 1. Per past discussions with the valve vendor, deviations within this range are within the design requirements of the valve and are not cause for concern over the condition of the valve. This conclusion is further supported by the fact that the valves demonstrated satisfactory test results after adjustment.

V. CORRECTIVE ACTIONS

Maintenance was performed on all three PSVs before retesting in order to restore them to required lift setpoints. On October 27-29, 1994, the required surveillances were completed satisfactorily, with all three PSVs being tested to state-of-the-art requirements identified through the WOG program.

VI. PREVIOUS SIMILAR EVENTS

There has been one previous similar event reported pursuant to 10CFR50.73. (LER 91-026 on Unit 1.) This previous event was a result of an inadequate test methodology being used to originally set the PSV lift setpoints. The original method used 200 degree (F) water whereas the improved method uses saturated steam. Unit 1 and Unit 2 PSVs are now tested with this improved method.

VII. ADDITION INFORMATION

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