Log # TXX-96509
File # 10200
Ref. # 10CFR50.73(a)(2)(vii)

C. Lance Terry
Group Vice President

November 14, 1996

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)-UNIT 1
DOCKET NOS. 50-445
INOPERABLE TRAIN/CHANNEL IN SAFETY RELATED SYSTEM
LICENSEE EVENT REPORT 445/96-008-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 96-008-00 for Comanche Peak Steam Electric Station Unit 1, "Two Pressurizer Safety Valves Found With Unsatisfactory Lift Setpoints."

Sincerely,

C. L. Terry

RTB:rtb
Enclosure

cc: Mr. L. J. Callan, Region IV
Mr. J. I. Tapia, Region IV
Resident Inspectors, CPSES

IE22

220069

Enclosure to TXX- 96509

NRC FORM 366 (4-95) — U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 4/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC. 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC. 20503.

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Title (4)

TWO PRESSURIZER SAFETY VALVES FOUND WITH UNSATISFACTORY LIFT SETPOINTS

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 Name	Docket Numbers
10	17	96	96	- 008	- 00	11	14	96	CPSES UNIT 2	05000446
									N/A	05000

Operating Mode (9)		This report is submitted pursuant to the requirements of 10 CFR §: (Check one or more) (11)			
Power Level (10)	0	20.2201 (b)	20.2203 (a) (2) (v)	50.73 (a) (2) (i)	50.73 (a) (2) (viii)
		20.2203 (a) (1)	20.2203 (a) (3) (i)	50.73 (a) (2) (ii)	50.73 (a) (2) (x)
		20.2203 (a) (2) (i)	20.2203 (a) (3) (ii)	50.73 (a) (2) (iii)	73.71
		20.2203 (a) (2) (ii)	20.2203 (a) (4)	50.73 (a) (2) (iv)	OTHER
		20.2203 (a) (2) (iii)	50.36 (c) (1)	50.73 (a) (2) (v)	Specify in Abstract below or in NRC Form 366A
		20.2203 (a) (2) (iv)	50.36 (c) (2)	X 50.73 (a) (2) (vii)	

Licensee Contact For This LER (12)

Name	Telephone Number (Include Area Code)
RAFAEL FLORES - SYSTEM ENGINEERING MANAGER	(817)897-5590

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
				N						

Supplemental Report Expected (14)				Month	Day	Year
YES (If yes, completed EXPECTED SUBMISSION DATE)	X	NO	EXPECTED SUBMISSION DATE (15)			

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On October 17, 1996, Comanche Peak Steam Electric Station (CPSES) Unit 1 was defueled in the 1RF05 refueling outage. The Pressurizer Safety Valves (PSVs) had been returned to the Westinghouse Western Service Center for surveillance testing as required by the CPSES Technical Specifications and ANSI/ASME OM-1987 Part I. Two of the PSVs as-found lift setpoints were measured to be unsatisfactory. These failures resulted in two independent trains being inoperable in a single safety related system designed to mitigate the consequences of an accident.

TU Electric believes that this event was caused as a result of setpoint drift of a magnitude within the design requirements of the valves. The valves were returned to operable status using normal rework for these components with no further actions planned.

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Text (if more space is required, use additional copies of NRC Form 366A) (17)

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 designed to mitigate the consequences of an accident.

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On October 17, 1996, Comanche Peak Steam Electric Station (CPSES) Unit 1 was defueled, 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

In October of 1996, all three Pressurizer Safety Valves (PSVs) had been returned to the Westinghouse Western Service Center for surveillance testing in support of the fifth refueling outage for CPSES Unit 1. Testing was performed using a procedure designed to comply with ANSI/ASME OM-1987 Part 1 and Westinghouse Owner's Group guidance, using saturated steam as the test medium. On October 17, 1996, PSV 1-8010A was found to be 1.4% 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 17, 1996. PSV 1-8010B lifted at 1.1% above the setpoint, the other PSV was found with an acceptable setpoint. Both PSVs were subsequently reworked and the lift set pressures were verified to be within Technical Specification limits.

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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.

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B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

The PSVs were initially set to within TS limits on March 31, 1995, and were considered operable until they were determined to be inoperable on October 17, 1996. Although the PSV lift set pressures 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 set pressure. 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 (EIIS:(RV)(AB)) or Steam Dump Valves (EIIS:(V)(SB)).

In the accident analyses, each of the three PSVs is assumed to open at its nominal set pressure, based on the expectation of a random distribution of as-found set pressures about the nominal set pressure. The CPSES Unit 1 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. 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 set pressure range. In conclusion, although these Unit 1 PSVs did not meet setpoint criteria required by CPSES Technical Specifications by a narrow margin, the functional capacity of these PSVs was not affected. No challenges to operate the PSVs were presented by the RCS and the health and safety of the public was unaffected.

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IV. CAUSE OF THE EVENT

A specific 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 both PSVs before retesting in order to restore them to required lift setpoints. On October 17, 1996, the required surveillances were completed satisfactorily, with all three PSVs being tested to state-of-the-art requirements identified through the Westinghouse Owners Group program.

VI. PREVIOUS SIMILAR EVENTS

A previous similar event has been reported for CPSES pursuant to 10CFR50.73, LER 446/96-018 on Unit 2. The previous event was a result of setpoint drift of a magnitude within the design requirements of the valves, similar to the current event.

VII. ADDITIONAL INFORMATION

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