TUELECTRIC

Log # TXX-94256
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Ref. # 10CFR50.73(a)(2)(i)

September 30, 1994

C. Lance Terry
Group Vice President

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) - UNIT 1
DOCKET NO. 50-445
CONDITIONS PROHIBITED BY TECHNICAL SPECIFICATION
LICENSEE EVENT REPORT 445/94-005-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 94-005-00 for Comanche Peak Steam Electric Station Unit 1, "Surveillance for Steam Generator Water Level and for Containment Pressure Channels were not Performed in 1992/1993 Due to Personnel Error".

Sincerely,

C. Lance Terry

OB:bm

Enclosure

cc: Mr. L. J. Callan, Region IV
Mr. D. D. Chamberlain, Region IV
Resident Inspectors, CPSES

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Enclosure to TXX-94256

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)

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

Title (4)

SURVEILLANCES FOR STEAM GENERATOR AND FOR CONTAINMENT PRESSURE CHANNELS WERE NOT PERFORMED IN 1992/1993 DUE TO PERSONNEL ERROR

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
									COMANCHE PEAK-UNIT 2	0|5|0|0|0|4|4|6
0|8	3|1	9|4	9|4	- 0|0|5 -	0|0	0|9	3|0	9|4	N/A	0|5|0|0|0| |

Operating Mode (9): 1

Power Level (10): 1\|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)	X 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
STEVE SMITH, WORK CONTROL MANAGER	8|1|7	8|9|7 - 6|0|7|0

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)

☐ 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 August 31, 1994, during a Quality Assurance audit it was discovered that Technical Specification surveillances were not performed on Unit 1 for the following transmitters: three of the Steam Generator Water Level Hi-Hi channels (EIIS)(SGLI)(SB)) and for Containment pressure (EIIS)(PI(VA)).

Technical Specifications require all instrument channels that monitor the Containment pressure and the Steam Generator (SG) Water Level have response time testing surveillances completed in 'n' times 18 months where 'n' equals the number of channels. Personnel responsible for this surveillance activity failed to properly schedule the surveillances based on the frequency and function. The corrective actions were to review the surveillance database and correct the frequency as required.

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

Any operation or condition prohibited by the Technical Specifications.

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

At the time of discovery on August 31, 1994, and on September 1, 1994, Comanche Peak Steam Electric Station (CPSES) was in Mode 1, Power Operation, with reactor power at 100 percent. Unit 2 was in MODE 1, Power Operation, at 88 percent and increasing reactor power to 100 percent.

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 to these events.

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

Event One - Containment Pressure

A Containment pressure Technical Specification surveillance was missed on Unit 1 in June 1992. Containment pressure is monitored by four instruments. Three instruments input into the Hi 1 signal, but all four input into the Hi 3 signal. The Technical Specifications require all channels to have response time testing completed in "n" times eighteen months where 'n' is the number of channels. Channel 1-P-0934 was completed August 2, 1990, which met the Technical Specifications requirement for both Hi 1 and Hi 3. Although a surveillance was performed for Hi 3 within eighteen months, the next Hi 1 channel, 1-P-0935 was not completed until December 4, 1992, which is five and one half months past its required surveillance date of June 19, 1992. To satisfy the Technical Specification for Hi 3, containment pressure, each of the four channels is required to be response time tested every seventy-two months, with at least one every eighteen months. However, the Hi 1 containment pressure requires each of the three channels to be response time tested every fifty-four months, with at least one every eighteen months.

Event Two - Steam Generator Water Level

Steam Generator (SG) Water Level Hi-Hi Technical Specifications surveillances were missed on Unit 1 for three Steam Generators (SG) in early 1992 or late 1993. SG Narrow Range Level is monitored by four instruments. Three instruments input into the Steam Generator Water Level Hi-Hi signal and four input into the Lo-Lo signal. The Technical Specifications require all channels to have response time testing completed in 'n' times eighteen months where 'n' equals the number of channels.

SG #1 had a surveillance performed on March 15, 1990, which satisfied the Technical Specifications for both Hi-Hi and Lo-Lo, but the next Hi-Hi response time surveillance was not performed until December 10, 1992, which is ten and one half months past its required surveillance date of January 31, 1992. The narrow range level that does not input into the Steam Generator water level Hi-Hi signal was tested on October 31, 1991, which met the Technical Specification requirement for Lo-Lo testing only.

SG #2 had a surveillance performed on November 1, 1991, which satisfied the Technical Specifications for both Hi-Hi and Lo-Lo, but the next Hi-Hi channel surveillance was performed on October 11, 1993, which is twenty three days past its required surveillance date of September 18, 1993. The narrow range level that does not input into the SG water level Hi-Hi signal was tested on December 10, 1991, which met the Technical Specification requirements for Lo-Lo testing only.

SG #3 had a surveillance performed on March 14, 1990, which satisfied the Technical Specifications for both Hi-Hi and Lo-Lo, but the next Hi-Hi channel surveillance was performed on December 10, 1992, which is ten and one half months past its required surveillance date of January 30, 1992. The narrow range level that does not input into the SG water level Hi-Hi signal was tested October 21, 1991, which met the Technical Specifications requirements for Lo-Lo testing only.

A review of the surveillance documents for the equipment concluded that at the present time all surveillances are current for both Units; and no equipment degradation was noted when the surveillances were performed during the previous refueling cycles.

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

Both events were discovered during a Quality Assurance audit by Nuclear Overview personnel (utility, non-licensed).

II. COMPONENT OR SYSTEM FAILURES

A. FAILED COMPONENT INFORMATION

Not applicable - There were no component failures associated with these events.

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

Not applicable - There were no component failures associated with these events.

C. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Not applicable - There were no component failures associated with these events.

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

Not applicable - There were no component failures associated with these events.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

Not applicable - There were no safety system responses associated with these events.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Not applicable - There were no safety systems rendered inoperable due to a failure.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The Hi-1 and Hi-2 containment pressure signals trip functions are credited in the analyses of the containment response to a steamline break inside containment, as described in FSAR Chapter 6. The high-high steam generator water level trip function is credited in the analysis of the Increase in Feedwater Flow event described in FSAR Section 15.1. The response times listed in the Technical Requirements Manual for these trip functions are explicitly assumed in the accident analyses. Based upon a review of the actual measured response times for these functions, there is significant margin between the measured response time and the allowable response time; hence, any small changes in the response time would not adversely impact the accident analysis assumptions. Furthermore, recent analyses performed by the Electric Power Research Institute (ref. EPRI NP-7423) and the Westinghouse Owners Group (ref. WCAP - 13632) demonstrated that any large changes in the sensor response time would be detected through normal channel calibrations which were performed during the previous refueling cycle. Gross failures in the sensors which could lead to changes in the response times could also be detected during the shiftly channel checks and during the quarterly channel operability tests.

Based on the available margins between the measured response times and the allowable response times, which can be used to offset any small changes in the response time, and the gross changes in the response time which would be detectable through normal, periodic surveillances which were performed, it is concluded that these events did not adversely impact the safe operation of Unit 1 and Unit 2 or the health and safety of the public.

IV. CAUSE OF THE EVENT

The individual who originally scheduled these activities failed to identify the unique nature of some of the instrument loops and that all of the instrument loops did not perform the same function. This failure resulted in the instrument loops being scheduled such that the Technical Specification frequency requirements for testing these instruments were not met. Five groups of instruments have one instrument loop which does not perform all the function. These include four steam generator water level groups and one containment pressure group. When the "fourth" procedure in each of these groups is performed, then one of the functions are not satisfied (i.e., steam generator level HI-HI signal and containment pressure high-1 signal).

V. CORRECTIVE ACTIONS

TU Electric's review concluded that the response time testing was not performed on the transmitters only. The process instrumentation and Solid State Protection System were properly tested.

Although, at the time of discovery, the Technical Specification Surveillances for containment pressure and steam generator water level were current, the following corrective actions were taken:

1. The Technical Specification surveillance database was reviewed to assure that other similar surveillance requirements have been met. No further violations due to variations between channels for the same instrument function were identified. Although the same scheduling problem was found in the Unit 2 Technical Specification surveillance database for similar surveillances, this problem will be corrected prior to the first required surveillances.

2. Sensor response time testing frequency has been corrected to account for the variations between the channels.

VI. PREVIOUS SIMILAR EVENTS

TU Electric has reviewed previous similar events. The root causes of the previous events are sufficiently different from this event and the corrective action for those events would not have prevented this event. TU Electric believes that the corrective actions for the previous events have been generally effective.