TU ELECTRIC

Log # TXX-90192
File # 10200
910.3
915.6
Ref. # 50.73
50.73(a)(2)(ii)

W. J. Cahill
Executive Vice President

June 22, 1990

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NO. 50-445
OPERATION PROHIBITED BY TECHNICAL SPECIFICATIONS
LICENSEE EVENT REPORT 90-016-00

Gentlemen:

Enclosed is Licensee Event Report 90-016-00 for Comanche Peak Steam Electric Station Unit 1, "Three of Four Steam Generator Atmospheric Relief Valves Inoperable Due to Insufficient Stroke Length Settings."

Sincerely,

William J Cahill Jr

William J. Cahill, Jr.

JRW/daj

Enclosure

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

IE22
1/1

Enclosure to TXX-90192

NRC FORM 366 — U.S. NUCLEAR REGULATORY COMMISSION

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 1 0

Title (4): THREE OF FOUR STEAM GENERATOR ATMOSPHERIC RELIEF VALVES INOPERABLE DUE TO INSUFFICIENT STROKE LENGTH SETTINGS

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
0 5	2 3	9 0	9 0	- 0 1 6 -	0 0	0 6	2 2	9 0	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

Operating Mode (9): 3

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
G. P. McGEE	SUPERVISOR, COMPLIANCE	Area Code 8 1 7 — 8 9 7 - 5 4 7 7

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
X	S B	R V	F 1 3 0	Y					

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 May 21, 1990, Comanche Peak Steam Electric Station Unit 1 was conducting capacity testing to verify that the Steam Generator Atmospheric Relief Valves (ARVs) stroked fully, using feedwater flow increase as a qualitative indicator of valve stroke. Following the test, Engineering determined that valves 1-PV-2325, 1-PV-2326 and 1-PV-2327 did not provide the minimum required steam flow capacity to support the Design Basis Accident Analysis. As a result, the three ARVs were declared inoperable resulting in Unit 1 entry into Technical Specification Limiting Condition for Operation (LCO) 3.0.3. The valves were subsequently calibrated and declared operable allowing exit from LCO 3.0.3.

The event resulted from two causes: 1) The pneumatic controls for three of the four ARVs had drifted out of calibration, and 2) the specified stroke length for two of these three ARVs was reduced due to an inadequate review and approval process for Instrumentation & Control (I&C) data calibration sheets. Corrective actions include setting the valves to the appropriate configuration and conducting an evaluation to establish the frequency for verifying ARV stroke length. Also, the calibration data sheets were corrected and the I&C program for the revision of calibration data sheets had previously been revised to require the Supervisor, I&C Engineering to review any changes to design requirements.

I. DESCRIPTION OF THE REPORTABLE EVENT

A. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

At 1930 on May 23, 1990, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 3, Hot Standby, with Reactor Coolant System (EIIS:(AB)) temperature and pressure at 557 degrees Fahrenheit and 2250 pounds per square inch-gage, respectively.

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

The reportable event included plant operation for approximately 64 days with three of four Steam Generator Atmospheric Relief Valves (ARVs) (EIIS:(RV)(SB)) unable to support the Design Basis Accident Analysis. Therefore, three ARVs were inoperable at the start of the event.

C. REPORTABLE EVENT CLASSIFICATION(S)

Any operation or condition prohibited by the plant's Technical Specifications.
Any event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis.

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

At 0945 on May 21, 1990, CPSES Unit 1 was in Mode 1, Power Operation, operating at 35 percent reactor power. Performance and Test engineers (contractor and utility, non-licensed) had received permission from Operations to begin a test of the ARVs. The test was intended to verify that the ARVs stroked fully, using feedwater flow increase as a qualitative indicator of stroke. Revision 0 of the ARV Capacity Test procedure contained "Acceptance Criteria" requiring that each valve will open and close under normal hot steam conditions. In addition, the test procedure contained "Review Criteria" requiring that each valve will have the capacity to pass steam flow equivalent to 2.5 (+1, -1) percent of the total rated steam flow. "Acceptance Criteria" provide observable results to be used to judge the acceptability (SAT or UNSAT) of

NRC FORM 366A	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED OMB NO. 3150-0104 EXPIRES: 4/30/92
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION		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)	LER Number (6) Year	Sequential Number	Revision Number	Page (3)
COMANCHE PEAK - UNIT 1	0 5 0 0 0 4 4 5	9 0	- 0 1 6	- 0 0	0 3 OF 1 0

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

the task while "Review Criteria" are utilized as an indicator of potential problems rather than as a limiting condition. "Review Criteria" do not necessarily render a test unsatisfactory if not achieved.

The test procedure was started at 0955 and completed at 1400. The measured capacities for ARVs 1-PV-2325, 2326, 2327 and 2328 were approximately 2.8, 2.4, 3.8, and 4.7 percent, respectively (as referenced to 100 percent steam flow, without pressure corrections). The deviation from the "Review Criteria" for the last two valves was documented and engineering was requested to evaluate the operability impact of the apparent excess capacity.

While Engineering performed their review, Instrument and Control (I&C) technicians (utility, non-licensed) were sent to measure the stroke lengths for each valve. The design specification for the stroke length was 1-3/8 (+1/16, -0) inches. The measured stroke lengths for 1-PV-2325 through 2328 were 1-3/32, 1-1/16, 1-5/16, and 1-3/8 inches, respectively. The differences in ARV capacity appeared to correlate with the differences in stroke length.

The Engineering review of the test results determined that valves 1-PV-2327 and 2328 did not have excess capacity. Instead, it was determined that with the exception of 1-PV-2328, the ARVs did not appear to provide the minimum required steam flow capacity to support the Design Basis Accident Analysis for the Steam Generator Tube Rupture (SGTR) event. At 1930 on May 23, the three ARVs were declared inoperable resulting in Unit 1 entry into Technical Specification Limiting Condition for Operation (LCO) 3.0.3 when the LCO for Technical Specification 3.7.1.7 (requires a minimum of two operable ARVs) could not be met. Concurrent with declaring the three ARVs inoperable, actions were taken to commence plant cooldown from Mode 3 to Mode 4, Hot Shutdown, as required by Technical Specification LCO 3.0.3. (Unit 1 had entered Mode 3 on May 22 following a planned reactor trip to perform a Loss of Offsite Power Test).

I&C completed the valve calibration (bench set) of 1-PV-2327 with the valve stroke length reset to 1-3/8 inches and at 2231 on May 23, Operations successfully completed the Technical Specification required surveillance test allowing 1-PV-2327 to be declared operable. This allowed Unit 1 to exit LCO 3.0.3 at 2236 and enter the 72 hour Action Statement for Technical Specification 3.7.1.7 (2 ARVs operable). At 2055 on May 23, the Nuclear Regulatory Commission (NRC) Operations C[illegible]ter was

NRC FORM 366A 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)
TEXT CONTINUATION

Facility Name (1)	Docket Number (2)	LER Number (6) Year	Sequential Number	Revision Number	Page (3)
COMANCHE PEAK - UNIT 1	0 5 0 0 0 4 4 5	9 0	0 1 6	0 0	0 4 OF 1 0

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

notified of the inoperable ARVs via the Emergency Notification System in accordance with 10CFR50.72(b)(2)(i). A followup call to the NRC was subsequently made at 2241 which identified that LCO 3.0.3 had been exited.

The valve calibration (bench set) and Technical Specification required surveillance testing for 1-PV-2326 and 1-PV-2325 were subsequently completed and the valves declared operable at 0510 on May 24 and at 1919 on May 25, respectively. The Action Statement for Technical Specification 3.7.1.7 was exited and a Tracking-Limiting Condition of Operation Action Requirement was initiated to ensure the ARV Capacity Test was performed when appropriate operating conditions were achieved. After resetting the ARV's stroke length to 1-3/8 inch and successfully completing the Technical Specification required surveillance testing of the three ARVs, the Operations Manager (utility, licensed) gave permission for Unit 1 entry into Mode 2, Startup, and Mode 1.

The ARV Capacity Test Procedure "Review Criteria" was changed to "Acceptance Criteria". The new "Acceptance Criteria" for ARV capacity was changed to require 779,000 to 968,000 pound-mass per hour (LBMH). On May 26, once necessary plant conditions were achieved, a second ARV Capacity Test was begun; however, it was only partially completed. The data was inconclusive as the test was interrupted by a reactor trip caused by a feedwater control valve (EIIS:(FCV)(SJ)) problem unrelated to the test.

On May 29, the ARV Capacity Test was reperformed. Measured flow rates were 779,000; 585,000; 710,000; and 668,000 LBMH. Based on an Engineering analysis, the ARVs were determined to be acceptable for the existing plant conditions (50 percent power or less).

During the period from May 30 to June 5, Unit 1 remained at 50 percent power while assessing operations (as committed to the NRC prior to increasing power to 75 percent). I&C checked the stroke lengths following the May 29 test performance and found the stroke of 1-PV-2326 (flowrate of 558,000 LBMH) to be less than specified (1-7/32 versus 1-3/8 inches) when operated from the control room. Previously, I&C had stroked the ARVs locally using instrument air when setting the mechanical travel stops at 1-3/8 inches. The control loop was recalibrated and the valve stroked to the correct length from the control room. All ARVs were verified to stroke the correct length from the control room. The valve vendor (Fisher Controls) was also brought on site to assist in valve calibration.

**LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION**

Facility Name (1)	Docket Number (2)	LER Number (6) Year	Sequential Number	Revision Number	Page (3)
COMANCHE PEAK - UNIT 1	0 5 0 0 0 4 4 5	9 0	- 0 1 6	- 0 0	0 5 OF 1 0

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

On June 7, Engineering re-established the design basis for the ARVs. The ARVs are required to remove a minimum of 756,000 LBMH at 1200 pounds per square inch-atmosphere (PSIA) to ensure that the capacity is sufficient to cool down the plant during a loss of off-site power such that at 100 PSIA the valves pass a minimum of 62,150 LBMH. This value (62,150 LBMH) bounds the capacity required for a SGTR event, such that there will be a sufficient cooldown rate to depressurize the reactor coolant system, allowing an operator to stop the leakage prior to the affected steam generator filling with liquid. In addition, the maximum relief capacity of each ARV is 968,400 LBMH so as not to exceed the maximum cooldown rate specified by the steam supply vendor. The bounding analysis for this maximum value is the stuck open main steam safety valve, per FSAR Chapter 15.1.5. The maximum value is also used in the radiological release calculations for determining the off-site dose consequences following an SGTR event.

Engineering obtained confirmatory information from the valve vendor stating that steam flow through the valves can be correlated to valve stroke within 3 percent. Thus, valve stroke length is the appropriate parameter to measure in determining valve operability, rather than using a secondary heat balance (due to the difficulty in accounting for the flows in all available steam flow paths). Engineering issued a design modification to increase valve stroke an additional 1/16 inch to 1 7/16 (+1/16, -0) inches. This was done to ensure that margin exists between the minimum and maximum capacity requirements and the actual valve setting. A comparison of steam flows calculated at the revised stroke lengths to the minimum and maximum required flow rates bounds the 3 percent tolerance with margin remaining. The design modification was implemented and the post-modification testing (verification of stroke length) and Technical Specification required surveillance testing were successfully completed. The ARVs were then declared fully operable (to 100 percent power) on June 14, 1990.

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

The inoperability of the three ARVs was discovered by the Engineering review of the test results for the ARV Capacity Test.

Enclosure to TXX-90192

NRC FORM 366A	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED OMB NO. 3150-0104 EXPIRES: 4/30/92
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION		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)	LER Number (6) Year	Sequential Number	Revision Number	Page (3)
COMANCHE PEAK - UNIT 1	0 5 0 0 0 4 4 5	9 0	- 0 1 6	- 0 0	0 6 OF 1 0

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

II. COMPONENT OR SYSTEM FAILURES

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

Three of the four ARVs were determined to be inoperable since the valves did not stroke sufficiently to support the Design Basis Accident Analysis. This resulted from the stroke length specified on the I&C data calibration sheets for two of the ARVs (1-PV-2325 and 1-PV-2326) being lower than the established design requirements combined with the fact that the pneumatic controls for 1-PV-2325, 1-PV-2326 and 1-PV-2327 had drifted out of calibration.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

The causes for the inoperable ARVs are described in Section IV.

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

Not applicable - Although the ARVs are containment isolation valves and provide a containment isolation function, the ability to perform this function was not affected by the problems identified in this event.

D. COMPONENT INFORMATION

ARVs 1-PV-2325, 2326, and 2327
Manufacturer: Fisher Controls
Model Number: 667-EWP

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

Not applicable - There were no manual or automatic safety system responses as a result of this event.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

The ARVs do not directly correlate with the term "safety system train." However, the ARVs do provide a safety function and were required to be operable in accordance with Technical Specifications beginning on March 20, 1990 when Mode 3 was entered (approximately 64 days passed until the inoperability was identified by ARV Capacity Testing). Upon discovery of the inoperability at 1930 on May 23, 1-PV-2327, 2326, 2325 were restored to an operable status within 3, 10, and 48 hours (approximate times), respectively.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

A minimum ARV capacity is required for the following reasons:

1) Sufficient ARV capacity is needed to assure a timely plant cooldown for the mitigation of the design basis SGTR accident. This cooldown is required in order to allow the primary-to-secondary leakage to be terminated by an operator prior to the affected steam generator (EIIS:(SG)(AB)) filling with liquid.

2) Sufficient ARV capacity is required to allow for cooling the plant from Hot Standby (Mode 3) to Residual Heat Removal (RHR) (EIIS:(BP)) system cut-in conditions in the event of a loss of offsite power. The cooldown must be accomplished prior to the depletion of Condensate Storage Tank (CST) (EIIS:(TK)(KA)) inventory.

An analysis was performed of the "as found" capacities during the initial performance of ARV Capacity Test and the results are summarized below:

1) In the absence of a single active failure, the reduced ARV capacity would not have prevented the ARVs from performing their intended safety function during the mitigation of a SGTR accident occurring while the plant was operating at or below 100 percent power. In addition, other systems, such as the radiation monitors (EIIS:(MON)(IL)) (providing for early detection of an SGTR) and the steam dumps (EIIS:(RV)(SB)) (providing an alternate heat removal path), were available for use in the mitigation of an SGTR event.

2) In the absence of a single active failure, the reduced ARV capacity would have been sufficient to allow the ARVs to perform their function of cooling the Reactor Coolant System to RHR system cut-in conditions prior to depleting the CST following a plant trip from power levels at or below 100 percent.

IV. CAUSE OF THE EVENT

The stroke length of three of the ARVs was found to be lower than design requirements resulting in their inoperability. This occurred due to the following two causes:

ROOT CAUSE-1

The pneumatic controls for ARVs 1-PV-2325, 1-PV-2326 and 1-PV-2327 had drifted out of calibration.

ROOT CAUSE - 2

Prior to October 1989, an inadequate review and approval process existed for I&C calibration data sheets. This process allowed a revision to the "stand alone" data sheets for 1-PV-2325 and 1-PV-2326 which reduced the specified stroke length from 1-3/8 inch to 1-1/4 inch.

CONTRIBUTING FACTOR

The analytical determination of the 1-3/8 inch stroke length (verified by the valve vendor) resulted in a steam flow very near the minimum value required by the accident analysis Based on the uncertainties associated with correlating valve flow capacity and valve stroke length, it would have been prudent to establish a valve stroke setting with adequate tolerances.

V. CORRECTIVE ACTIONS

A. CORRECTIVE ACTIONS TO PREVENT RECURRENCE

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

Facility Name (1)	Docket Number (2)	LER Number (6) Year	Sequential Number	Revision Number	Page (3)
COMANCHE PEAK - UNIT 1	0 5 0 0 0 4 4 5	9 0	- 0 1 6 -	0 0	0 9 OF 1 0

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

ROOT CAUSE - 1

The pneumatic controls for ARVs 1-PV-2325, 1-PV-2326 and 1-PV-2327 had drifted out of calibration.

CORRECTIVE ACTION

An evaluation will be performed to establish the frequency for verifying ARV stroke length to ensure that the valves are within tolerance. Valve stroke measurement will be performed after opening the valve from the control room via the Manual/Auto station.

ROOT CAUSE - 2

Prior to October 1989, an inadequate review and approval process existed for calibration data sheets.

CORRECTIVE ACTION

The I&C program for revision of calibration data sheets was revised in October 1989 to require the Supervisor, I&C Engineering (in lieu of a Shop Supervisor) to review any changes to design requirements. This program change is adequate for control of future revisions of calibration data sheets.

CONTRIBUTING FACTOR

The analytical determination of 1-3/8 inch stroke length resulted in a steam flow rate very near the minimum value required by the accident analysis.

CORRECTIVE ACTION

The appropriate calculation has been revised and a design modification has been implemented to increase valve stroke an additional 1/16 inch. This was done to ensure margin exists between the minimum and maximum capacity requirements and the actual valve settings.

**LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION**

Facility Name (1)	Docket Number (2)	LER Number (6) Year	Sequential Number	Revision Number	Page (3)
COMANCHE PEAK - UNIT 1	0 5 0 0 0 4 4 5	9 0	- 0 1 6 -	0 0	1 0 OF 1 0

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

B. CORRECTIVE ACTION TAKEN ON GENERIC CONCERNS IDENTIFIED AS A DIRECT RESULT OF THE EVENT

GENERIC IMPLICATION

Since a backfit review was not required as part of the I&C review and approval process change in October 1989, additional deficiencies may exist for "stand alone" data calibration forms.

CORRECTIVE ACTION

To provide additional assurance that similar deficiencies do not exist, a representative sample of quality related "stand alone" calibration data sheets was reviewed to assess consistency with design requirements. Unlike the "stand alone" data calibration sheets which provide the sole source of project documentation for the calibration data, data calibration sheets with supporting procedures, documentation, etc. are considered validated and there is no need to include them in the sample. No deficiencies were found during this evaluation.

VI. PREVIOUS SIMILAR EVENTS

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

VII. ADDITIONAL INFORMATION

All times are approximate and Central Daylight Savings Time (CDT).