TU ELECTRIC

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November 11, 1992

William J. Cahill, Jr.
Group Vice President

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE
LICENSEE EVENT REPORT 92-022-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 92-022-00 for Comanche Peak Steam Electric Station Unit 1, "Manual Reactor Trip Due to Feedwater Flow Control Valve Failure".

Sincerely,

William J. Cahill, Jr.

William J. Cahill, Jr.

JET/tg

Enclosure

c - Mr. J. L. Milhoan, Region IV
Resident Inspectors, CPSES (2)

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
COMANCHE PEAK-UNIT 1	05000445	1 OF 06

Title (4): **MANUAL REACTOR TRIP DUE TO FEEDWATER FLOW CONTROL VALVE FAILURE**

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
10	12	92	92	- 022	- 00	11	11	92	N/A	05000
									N/A	05000

Operating Mode (9): 1

Power Level (10): 100

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

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

Licensee Contact For This LER (12)

Name	Area Code	Telephone Number
D. E. BUSCHBAUM, COMPLIANCE SUPERVISOR	817	897-5851

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
A	FW	OPVI	0088	Y					

Supplemental Report Expected (14)

- [ ] Yes (If yes, complete Expected Submission Date)
- [x] No

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

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

At 1210 on October 12, 1992, Steam Generator (SG) Number (No.) 4 Steam and Feedwater Flow Mismatch Alarm annunciated. The Reactor Operator (RO) immediately took manual control of the SG No. 4 Feedwater Flow Control Valve (FCV). The SG No. 4 FCV had failed in the closed direction, causing loss of feedwater flow to SG No. 4, and would not respond to manual control. At 1214, with SG No. 4 level at 28 percent and dropping, the RO manually tripped the reactor. Plant systems responded as expected. The plant was stabilized in Mode 3, Hot Standby, at approximately 1220 on October 12, 1992.

The root cause of this event was the failure of a spring in the SG No. 4 FCV pressure regulator due to the wrong pressure regulator being installed. Corrective actions include repair of SG No. 4 FCV, and a revision to the Master Parts List. Generic corrective actions include verification of the correct regulators in the other FCVs and other non-safety critical valves and a separate review of the Master Parts List.

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 2	0 2 2	0 0	0 2 OF 0 6

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

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)(EIIS:(JC)).

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On October 12, 1992, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, with reactor power at 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 directly to the event.

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

At 1210 on October 12, 1992, Steam Generator (SG)(EIIS:(SG)(AB)) Number (No.) 4 Steam and Feedwater Flow Mismatch Alarm annunciated on the Main Control Board (MCBD)(EIIS:(MCBD)(IB)). The Balance of Plant (BOP) Reactor Operator (RO) (utility, licensed) noted that feedwater flow to SG No. 4 was less than steam flow and immediately took manual control of the SG No. 4 Feedwater Flow Control Valve (FCV)(EIIS:(FCV)(SJ)). The SG No. 4 FCV had failed in the closed direction, causing loss of feedwater flow to SG No. 4, and would not respond to manual control. At 1211, with SG No. 4 level dropping, the Main Feedwater Pumps (EIIS:(P)(SJ)) were manually increased to their maximum output and the SG No. 4 Feedwater Flow Control Bypass Valve (EIIS:(FCV)(SJ)) manually opened in an attempt to maintain SG No. 4 level. At 1214, with SG No. 4 level at 28 percent and dropping, the Reactor Operator (RO) (utility, licensed) manually tripped the reactor. Both Motor Driven Auxiliary Feedwater Pumps (EIIS:(P)(BA)) automatically started, restoring SG levels. Plant systems responded as expected. The plant was stabilized in Mode 3, Hot Standby, at approximately 1220 on October 12, 1992.

An event or condition that results in an automatic or manual actuation of any ESF, including the RPS, is reportable within 4 hours under 10CFR50.72(b)(2)(ii). At 1452 on October 12, 1992, the Nuclear Regulatory Commission Operations Center was notified of the event via the Emergency Notification System.

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

An alarm on the MCBD alerted the BOP RO to the situation. The BOP RO and RO took immediate corrective actions but could not maintain SG No. 4 level. The decision was made to manually trip the reactor prior to the automatic reactor trip on LO-LO SG Level.

II. COMPONENT OR SYSTEM FAILURES

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

A failure of the spring in the pressure regulator for SG No. 4 FCV caused the valve to fail closed and prevented control from the MCBD.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Examination of the failed SG No. 4 FCV pressure regulator revealed that an incorrect regulator had been installed. A pressure regulator with a range of 0-60 pounds per square inch (psig) was installed where a 0-125 psig regulator was required.

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

Not applicable - no failures of components with multiple functions have been identified.

D. FAILED COMPONENT INFORMATION

Pressure Regulator

1-FCV-540 PR1
Manufacturer: ITT, Hammel DAHL Conoflow
Model Number: FH-60 Airpac

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	05000445	92 - 022 - 00	04 OF 06

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

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The following safety system actuation signals occurred as a result of this event:

Reactor Protection System
Auxiliary Feedwater System (AFW)(EIIS:(BA))

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

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

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

This event is completely bounded by the accident analysis in Chapter 15.2.7 of the Final Safety Analysis Report for a Loss of Normal Feedwater. A reactor trip on low SG water level in any SG and the initiation of AFW provides the necessary heat removal capability. The RO manually tripped the reactor before the automatic trip for LO-LO SG Level was reached and the AFW flow initiated as required. Therefore, this event did not adversely affect the safe operation of CPSES Unit 1 or the health and safety of the public.

IV. CAUSE OF THE EVENT

ROOT CAUSE

The cause of the event was the failure of a spring in the SG No. 4 FCV pressure regulator. However, the root cause of this event was the installation of a 0-60 psig pressure regulator where a 0-125 psig pressure regulator was required. The 0-60 psig pressure regulator was listed as an acceptable replacement part in the Master Parts List. In September, 1990, a cut in the diaphragm of SG No. 4 FCV pressure regulator was discussed. Instrumentation and Control (I&C) (utility, nonlicensed) personnel went to the Warehouse to get replacement parts for the FCV. They received two pressure regulators, one 0-60 psig and one 0-125 psig; both listed as acceptable replacements for the FCV pressure regulators. I&C personnel installed the 0-60 psig pressure regulator into SG No. 4 FCV and returned the 0-125 psig pressure regulator to the Warehouse.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

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	05000445	92	022	00	05 OF 06

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

V. CORRECTIVE ACTIONS

A. CORRECTIVE ACTIONS TO PREVENT RECURRENCE

IMMEDIATE CAUSE

Failure of a spring in the SG No. 4 FCV pressure regulator.

CORRECTIVE ACTION

The pressure regulator for SG No. 4 FCV was replaced with a 0-125 psig regulator, as required, and the valve returned to service.

ROOT CAUSE

The root cause of this event was the installation of the wrong FCV pressure regulator.

CORRECTIVE ACTION

The Master Parts List, showing the 0-60 psig pressure regulators as an acceptable replacement part for the FCVs, has been revised to the required 0-125 psig pressure regulators.

The root cause of this event occurred in 1990. Recent corrective actions emphasing the importance of Self Verification and the need to reduce personnel errors at CPSES addresses the personnel error involved in this LER.

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

GENERIC CONCERN

The possibility exists that incorrect pressure regulators could be installed in the other three FCVs in Unit 1 and the four FCVs in Unit 2.

CORRECTIVE ACTION

The pressure regulators in the other three Unit 1 FCVs and the four Unit 2 FCVs were verified to be the 0-125 psig regulators, as required.

GENERIC CONCERN

The possibility exists that incorrect pressure regulators were installed in other non-safety critical valves in Unit 1.

CORRECTIVE ACTION

The Single Point Failure Analysis was reviewed to identify the non-safety critical valves. The pressure regulators were then identified for these valves. These regulators were then walked down and/or verified to be correct.

GENERIC CONCERN

The possibility exists that the Master Parts List could reference incorrect replacement parts for other components.

CORRECTIVE ACTION

A review of this issue has been previously initiated and will be formally documented as a Quality Assurance Deficiency in accordance with applicable CPSES Procedures.

VI. PREVIOUS SIMILAR EVENTS

Seven CPSES Licensee Event Reports (LER) describe previous events involving a reactor trip and the Feedwater System (EIIS:(SJ)). However, the root causes of these events and the resultant corrective actions are sufficiently different from this event to conclude that the previous corrective actions could not have been expected to prevent this event. Therefore, no previous similar events have been reported pursuant to 10CFR50.73.

VII. ADDITIONAL INFORMATION

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