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William J. Cahill, Jr.
Executive Vice President

October 9, 1990

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NO. 50-445
MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE
LICENSEE EVENT REPORT 90-027-00

Gentlemen:

Enclosed is Licensee Event Report 90-027-00 for Comanche Peak Steam Electric Station Unit 1, "Manual Reactor Trip Due to Shearing of Feedwater Flow Control Valve Feedback Linkage Arm."

Sincerely,

For William J. Cahill, Jr.

DEN/daj

Enclosure

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

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Enclosure to TXX-90327

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)

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Title (4) MANUAL REACTOR TRIP DUE TO SHEARING OF FEEDWATER FLOW CONTROL VALVE FEEDBACK LINKAGE ARM

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 9	0 7	9 0	9 0	- 0 2 7 -	0 0	1 0	0 9	9 0	N/A	0 5 0 0 0
									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)	☐ 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 J	F C V	C 6 3 5	YES					

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 September 7, 1990, at 0033, Comanche Peak Steam Electric Station (CPSES) Unit 1 Steam Generator (SG) Number (No.) 2 Feedwater Flow Control Valve (FCV) failed full open due to shearing of the positioner feedback linkage arm. The failed valve overfed SG No. 2 and the reactor was manually tripped at 0034 with SG No. 2 level at approximately 80 percent narrow range indicated level.

The plant was stabilized at 0043 in Mode 3. At 0130, a Balance of Plant Reactor Operator (RO) decreased auxiliary feedwater flow to SG No. 4 since the level was increasing faster than the other three. At 0232, a Relief RO noted that SG No. 4 level was approaching the Lo-Lo Level setpoint and increased auxiliary feedwater flow. The low level combined with the increased flow which caused a "shrink and swell" effect in SG No. 4 resulted in a Lo-Lo Level signal which generated an automatic start signal for the Auxiliary Feedwater System.

The cause of the linkage arm failure is attributed to fatigue resulting from flow induced oscillations. Corrective actions include repair of SG No. 2 FCV and a design modification to modify the valve internals to reduce flow induced oscillations.

**LICENSEE EVENT REPORT (LER)
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Text (If more space is required, use additional NRC Form 366A's) (17)

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

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

B. PLANT OPERATING CONDITIONS BEFORE THE EVENT

On September 7, 1990, at 0030, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, at 100 percent power.

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

No applicable - no structures, systems, or components were inoperable at the start of the event that contributed to the event.

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

On September 7, 1990, at 0033, Steam Generator (SG) (EIIS:(SG)(AB)) Number (No.) 2 Steam and Feedwater Flow Mismatch Alarm annunciated in the Control Room. A Reactor Operator (RO) (utility, licensed) noted that SG No. 2 Feedwater Flow Control Valve (FCV) (EIIS:(FCV)(SJ)) was at 45 percent demand; the valve is normally at 75 percent demand at full power. The RO took manual control of the FCV at the Main Control Board (MCB)(EIIS:(MCBD)(IB)) to reduce feedwater flow to SG No. 2, but the FCV did not respond. The RO informed the Shift Supervisor (utility, licensed) that the FCV had failed open and recommended the reactor (EIIS:(RCT)(AB)) be tripped since SG No. 2 level was 79 percent and increasing. The reactor was manually tripped at 0034 with SG No. 2 level narrow range at approximately 80 percent. After the manual reactor trip, the Main Feedwater Pumps (EIIS:(P)(BA)) tripped and Lo-Lo Level signals were received at 0034 on SG Nos. 1, 2, 3 and 4 which resulted in the automatic start of the Auxiliary Feedwater (AFW) Pumps (EIIS:(P)(BA)).

The operators responded to the reactor trip in accordance with Emergency Operating Procedures (EOP). The plant was stabilized by 0043 in Mode 3. To minimize Reactor Coolant System (RCS) (EIIS:(AB)) cooldown, AFW flow was controlled manually to maintain SG levels between 5 to 50 percent in accordance with the operation band provided in the EOP. At 0130, the Balance of Plant RO noticed that SG No. 4 level was increasing faster than the other three SGs and decreased the AFW flow to SG No. 4. SG level was checked periodically. At 0233, a Relief RO noted that SG No. 4 level was approaching the Lo-Lo Level setpoint and increased AFW flow. The increased flow caused a "shrink and swell" effect in SG No. 4. At 0234, a Lo-Lo Level signal occurred which resulted in a feedwater isolation signal. SG No. 4 level was at approximately 28 percent according to plant data at the time the Relief RO increased AFW flow; AFW automatic isolation signal setpoint is 28 percent SG level. Level was restored above the Lo-Lo-Level setpoint and equipment was restored to normal.

An event or condition that results in manual or automatic actuation of any ESF, including the RPS, is reportable within 4 hours under 10CFR50.72(b)(2)(ii). At 0111 on September 7, 1990, the Nuclear Regulatory Commission Operations Center (NRCOC) was notified of the manual reactor trip via the Emergency Notification System. At 0602, a follow-up notification to the NRCOC was made concerning the ESF actuation on SG No. 4 Lo-Lo Level.

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

The failure of the FCV was discovered by the RO when the valve did not respond to manual control.

II. COMPONENT OR SYSTEM FAILURES

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

The SG No. 2 FCV failed full open due to the positioner feedback linkage arm shearing causing the valve to overfeed SG No. 2.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
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B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

The immediate cause of the SG No. 2 FCV failing open was due to the shearing of the positioner feedback linkage arm. Based on a visual inspection, the failure is attributed to fatigue resulting from flow induced oscillations.

After the feedback linkage arm sheared, the positioner drive arm was then driven by the positioning spring to the valve closed position. The positioner responded to the erroneous closure signal by porting full output pressure to the valve diaphragm, causing the valve to go full open. After the manual reactor trip which initiated a feedwater isolation signal, redundant solenoid valves (EIIS:(FSV)(SJ)) between the

positioner and the valve diaphragm de-energized to vent the diaphragm and close the valve.

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 involved.

D. FAILED COMPONENT INFORMATION

FCV Actuator
Tag Number: 1-FCV-0520AO
Manufacturer: Copes-Vulcan Division
Model Number: D-100-160

Positioner
Manufacturer: Bailey Controls
Model Number: 5221030-8

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

1. The following safety systems actuated automatically as a result of the manual reactor trip. The appropriate components within these systems operated as designed.

   Feedwater Isolation Valves (EIIS:(ISU)(SJ))
   Turbine Generator Trip System (EIIS:(JJ))
   RPS

2. The following safety systems actuated on Lo-Lo Level signals received from SG Nos. 1, 2, 3, and 4 which occurred immediately after the manual reactor trip. The appropriate components operated as designed.

   AFW Pumps
   Condensate Makeup/Reject Valves (EIIS:(V)(SD))
   SG Blowdown System (EIIS:(WI))
   Process Sampling System (EIIS:(KN))

3. The following safety systems actuated automatically as a result of the ESF actuation on Lo-Lo Level in SG No. 4 which occurred at 0234.

   Condensate Makeup/Reject Valves (which had been reset from the previous actuation)
   AFW Pump FCVs (EIIS:(V)(BA)) (tripped from manual to automatic control which drove the valves fully open)

   The following systems were in their actuated state prior to the ESF signal at 0234.

   AFW Pumps
   SG Blowdown System
   Process Sampling System
   Turbine Generator Trip System
   RPS

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

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

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The manual reactor trip event on September 7, 1990, is bounded by the analysis presented in Final Safety Analysis Report (FSAR) Section 15.1.2, "Feedwater System Malfunctions that Result in an Increase in Feedwater Flow". The analysis assumes two FCVs fail open resulting in a step increase to 152 percent of nominal feedwater flow to each of the two affected SGs. This excessive flow analysis for the two SGs bounds a single feedwater malfunction event in which one FCV malfunctions, and results in a flow rate of 220 percent of nominal feedwater flow to one SG. The 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 reactor trip is attributed to the failure of SG No. 2 FCV. The mechanism of the failure is discussed in Section II.

V. CORRECTIVE ACTIONS

A. IMMEDIATE

Corrective Work Order was initiated to replace the positioners/controller feedback linkage arm on SG No. 2 FCV. An inspection was performed on the positioner. The following items were found:

1. The top ball and socket bearing was seized;

2. The positioner arm had a worn flat spot on the outer edge of the cam; and

3. The positioner cam follower/roller had a groove worn around the circumference of the roller.

The positioner and feedback linkage arm was replaced.

B. CORRECTIVE ACTIONS TO PREVENT RECURRENCE

Root Cause

Shearing of the feedback linkage arm was due to fatigue failure resulting from flow induced valve oscillations.

Corrective Action

1. The positioner and feedback linkage arm has been replaced.

2. A design modification (DM) has been approved which will modify the valve internals to reduce flow induced oscillation of the FCV. As a temporary measure, the valve has been repacked to reduce oscillation.

3. A preventive maintenance (PM) activity has been added to inspect and lubricate the FCV positioner at power on a monthly basis.

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

Generic Considerations

A similar failure may occur in the other three FCVs.

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Corrective Action-1

An inspection was performed on the other three FCVs. The items found include:

1. The top ball and socket bearing was seized;

2. The positioner arm had a worn flat spot on the outer edge of the cam; and

3. The positioner cam follower/roller had a groove worn around the circumference of the roller.

In addition, SG No. 4 FCV feedback linkage arm screw was worn half-way through. The above items are attributed to flow induced oscillations.

The valves have been repacked. Corrective Work Orders were initiated to repair and/or replace positioners and feedback linkage arms on the other three valves. Work was completed September 8, 1990.

The DM and PM activities identified as corrective actions for SG No. 2 FCV will be implemented on the other three valves.

Generic Considerations - 2

The possibility exists that the air operated control valves with Bailey positioners in the other systems may be subject to flow induced oscillations.

Corrective Action - 2

Air operated control valves with Bailey positioners are being inspected for similar positioner problems. The Bailey positioners in the plant primary systems do not perform a safety-related function. No problems to date have been identified. Any identified problems will be addressed on a case by case basis.

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VI. PREVIOUS SIMILAR EVENTS

The reactor trip on August 25, 1990, discussed in LER 90-025-00 was due to the failure of SG No. 2 FCV resulting from flow induced oscillations. The DM identified as corrective action to LER 90-025-00 would have prevented this event, but has not been implemented due to availability of parts by vendor.

V. ADDITIONAL INFORMATION

The times listed in the report are approximate and Central Daylight Savings Time (CDT).