TU ELECTRIC

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W. J. Cahill
Executive Vice President

June 26, 1990

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NO. 50-445
ENGINEERED SAFETY FEATURE ACTUATION
LICENSEE EVENT REPORT 90-017-00

Gentlemen:

Enclosed is Licensee Event Report 90-017-00 for Comanche Peak Steam Electric Station Unit 1, "Reactor Trip Due to Feedwater Control Valve Solenoid Failure."

Sincerely,

William J. Cahill, Jr.

FSP/daj

Enclosure

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

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/1

Enclosure to TXX-90213

NRC FORM 366 — U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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.

Facility Name (1): COMANCHE PEAK - UNIT 1
Docket Number (2): 0 5 0 0 0 4 4 5
Page (3): 1 OF 0 7

Title (4): REACTOR TRIP DUE TO FEEDWATER CONTROL VALVE SOLENOID 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
0 5	2 7	9 0	9 0	- 0 1 7 -	0 0	0 6	2 6	9 0	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

Operating Mode (9): 1
Power Level (10): 0 4 3

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: G. P. McGEE SUPERVISOR, COMPLIANCE
Telephone Number: Area Code 8 1 7 8 9 7 - 5 8 8 9

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	C L	A 6 1 0	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)

On May 27, 1990, at 0126 while performing Steam Generator Atmospheric Relief Valve (ARV) capacity testing, a Main Feedwater Flow Control Valve (FCV) failed closed. This resulted in reduced feedwater flow and decreasing Steam Generator (SG) No. 3 water level. The Operator closed the ARV, which was open for test purposes, and started to manually ramp down the main turbine to reduce reactor power. The Operator then opened the bypass flow control valve around the failed closed FCV, but the SG water level continued to decrease. At 0128, when No. 3 SG water level reached approximately 30 percent (automatic reactor trip is at 28 percent SG water level), the operator manually tripped the reactor. All other plant systems operated properly.

The cause of the event was the failure of a solenoid valve coil, associated with No. 3 SG FCV, due to rain water intrusion (FCV's are located outside). A temporarily removed cover allowed water to enter a junction box then drain via conduit to the solenoid coil housing.

Corrective action included the replacement of the failed solenoid coil and inspection of the other solenoids for water/moisture intrusion. An evaluation will determine if additional critical components exist in a similar configuration. Guidance for the conduct of outdoor maintenance activities will be addressed programatically.

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

I. DESCRIPTION OF THE REPORTABLE EVENT

A. PLANT OPERATING CONDITIONS BEFORE THE EVENT

On May 27, 1990 at 0126, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, with reactor power at 43 percent.

B. 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).

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

Not 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

At 0126 on May 27, 1990, the Steam Generator Atmospheric Relief Valve (ARV) (EIIS:(RV)(SB)) capacity testing was in progress on Steam Generator (SG) (EIIS:(SG)(AB)) No. 4 when the SG No. 3 Feedwater Flow/Steam Flow Mismatch Alarm (EIIS:(ALM)(IB)) annunciated. The Reactor Operator (utility-licensed) observed the Main Feedwater Flow Control Valve (FCV)(EIIS:(FCV)(SJ)) to SG No. 3 indicated fully closed with a 100 percent open demand on the controller. A few seconds later, a SG No. 3 Low Level Alarm (EIIS:(ALM)(IB)) annunciated. The Unit Supervisor (utility-licensed) ordered the SG No. 4 ARV closed and the Main Feedwater Flow Control Bypass Valve (EIIS:(FCV)(SJ)) to SG No. 3 opened. The Flow Control Bypass Valve was opened fully to increase feedwater flow to SG No. 3. At this time SG No. 3 level was at approximately 35 percent. The Balance of Plant Operator (utility-licensed) reduced load on the main turbine (EIIS:(TRB)(TA)) to attempt to reduce steam flow to less than feedwater flow on SG No. 3. A discussion between the Unit Supervisor and the Reactor Operator followed and a decision was made to trip the reactor (EIIS:(RCT)(AB)) if SG level could not be stabilized above 30

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percent on narrow range indication. At 0128, when SG level decreased to approximately 30 percent narrow range the Unit Supervisor ordered the Reactor Operator to manually trip the reactor. The reactor was tripped, all rods (EIIS:(ROD)(AA)) fully inserted into the core. Steam dumps (EIIS:(RV)(SB)) operated normally. An Auxiliary Feedwater System (EIIS:(BA)) actuation occurred as a result of Low Low SG Level Signal, and all components functioned as designed. All other plant systems operated properly. The plant was stabilized in Mode 3, Hot Standby.

An intermittent ground on Direct Current (DC) Bus 1ED2 (EIIS:(JA)(EJ)) was noticed the day before the event. This intermittent ground was never indicated for more than 10 seconds. Control Room personnel had reviewed the drawings for DC Bus 1ED2 before the event; however, the ground could not be located since the alarm was intermittent. Also, the loads of DC Bus 1ED2 are not conducive to isolation as they feed protection/control related equipment and to isolate them in Mode 1 would cause a reactor trip. On May 27, 1990, after the reactor trip, the ground indication was constant.

An event or condition that results in a manual or automatic actuation of any ESF, including the RPS is reportable within 4 hours under 10CFR50.72(b)(2)(ii). At approximately 0201 on May 27, 1990, 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**

The closure of SG No. 3 FCV was initially discovered as a result of a Feedwater Flow/Steam Flow Mismatch Alarm annunciation in the control room. Additionally, intermittent ground alarms were received on DC Bus 1ED2. A Work Order was subsequently initiated to troubleshoot the ground on DC Bus 1ED2, which disclosed water in the solenoid coil housing and associated conduit.

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II. COMPONENT OR SYSTEM FAILURES

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

The FCV closed because its position controlling solenoid had failed due to electrical grounding caused by water intrusion, resulting in loss of feedwater flow to SG No. 3.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Water intrusion at the junction box (EIIS:(JBX)(SJ)), creating partial submergence of the solenoid coil, has been determined to be the cause of the failure. When the solenoid was disassembled the coil (EIIS:(CL)(SJ)) was discovered to be sitting in approximately one inch of water.

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

1-FCV-0530-SV1 Solenoid Valve
Manufacturer: ASCO Valves, Automatic Switch Co.
Model Number: 208-492-1W (Solenoid Valve Coil)

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The following safety system actuations occurred as a result of the event:

Reactor Protection System (EIIS:(JC))

Auxiliary Feedwater System (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**

The Main Feedwater System is designed to provide feedwater to the SGs. The Main Feedwater FCV's regulate feedwater flow to the SGs. Failure of the solenoid coil causes the FCV to close (Fail Safe Position), restricting/isolating feedwater flow to the SGs.

If the event had occurred at full power an automatic reactor trip would have occurred. However, since the plant was operating at reduced power for testing, the operator was able to shutdown the reactor manually.

This event is bounded by the Final Safety Analysis Report Accident Analysis (Section 15.2.7) regarding a Loss of Normal Feedwater, which assumes the worst single failure in the Auxiliary Feedwater System. However, in this event, an Auxiliary Feedwater System actuation occurred and all components functioned as designed. 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

Water intrusion into SG No. 3 Main Feedwater Flow Control Valve 1-FCV-0530 Train "B" solenoid assembly (EIIS:(SOL)(SJ)) and associated conduit caused the solenoid coil to ground. Conduit from the solenoid assembly (approximately six feet long), connects to the bottom of the junction box. The configuration is installed with the junction box slightly elevated with respect to the solenoid assembly. Water entering the junction box drained via the conduit into the solenoid assembly, leading to the failure of the coil. It is concluded that the water intrusion resulted from maintenance activities which left the junction box cover

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temporarily removed during a period of heavy rainfall. This in turn failed the Flow Control Valve closed and restricted/shutoff feedwater flow to SG No. 3.

V. **CORRECTIVE ACTIONS**

A. **IMMEDIATE**

The junction box, conduit and solenoid valve housing were cleaned and dried. A new solenoid coil was installed.

B. **CORRECTIVE ACTIONS TO PREVENT RECURRENCE**

Root Cause

Failure of SG No. 3 Main Feedwater FCV Train 'B' solenoid coil was caused by water intrusion resulting from outdoor maintenance activities during a period of heavy rainfall.

Corrective Action

1. All work organizations will review the event with personnel to stress the need to adequately protect equipment from external environmental conditions during ongoing work activities.

2. All work organizations will evaluate their programs to ensure that appropriate guidance is provided prior to performing work on outdoor equipment.

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

Generic Considerations

Other solenoid assemblies on the FCV's may have had moisture/water intrusion.

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Corrective Action

The Work Order w... ...ised to inspect all the solenoid assemblies on the other Main Feedwater FCV' , a tot... ...f seven. All the solenoid coils were found dry with no signs of previou... wate... ...trusion. The Work Order also inspected for any other possible mecns ... ...ter intrusion into the solenoid assemblies. No additional intrusion paths we... ...dentified. Based on the inspection of the other solenoid assemblies this incident was determined to be an isolated case.

Generic Considerations

Additional critical components may exist which have the potential for a similar occurrence.

Corrective Action

The Single Point Failure Analysis identifies critical components whose failure can initiate a sequence of events, resulting in a reactor trip. This analysis will be reviewed to identify outdoor components which have the potential for a similar occurrence as the FCV solenoid valve coil. Inspections will be performed on any identified components.

VI. PREVIOUS SIMILAR EVENTS

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

VII. ADDITIONAL INFORMATION

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