TU ELECTRIC

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August 20, 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
INOPERABLE TRAIN/CHANNEL IN SAFETY RELATED SYSTEM
LICENSEE EVENT REPORT 92-020-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 92-020-00 for Comanche Peak Steam Electric Station Unit 1, "Motor Driven Auxiliary Feedwater Pump Test Line Isolation Valve Mispositioned Due to Valve Operating Apparatus Design Program".

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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TXX-92372

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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Title (4): MOTOR DRIVEN AUXILIARY FEEDWATER PUMP TEST LINE ISOLATION VALVE MISPOSITIONED DUE TO VALVE OPERATING APPARATUS DESIGN PROBLEM

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 Number(s)
07	21	92	92	- 020 -	00	08	20	92	N/A	05000
									N/A	05000

Operating Mode (9): 3
Power Level (10): 000

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)
- [ ] 50.73(a)(2)(iv)
- [ ] 50.73(a)(2)(v)
- [x] 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
				N					

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 1900 on July 20, 1992, the Reactor Operator (RO) noticed the output of the Motor Driven Auxiliary Feedwater Pump (MDAFWP) 1-02 was 100 gallons per minute higher than the combined flow to steam generator (SG) -03 and 04. The RO anticipated the flow mismatch was the result of flow transmitter problems. At approximately 1900 on July 21, 1992, with the flow transmitters verified to be operating correctly, the RO dispatched an Auxiliary Operator to check on the status of the MDAFWP 1-02 test line isolation valve (1AF-0055). The valve was found three-eighths of a turn open. The valve was closed and the flow mismatch eliminated.

The root cause of this event was a problem not anticipated with the design of the operating apparatus of 1AF-0055. The design did not allow the Operator to positively verify that the valve was closed. Corrective actions include a change to procedure providing positive verification of valve closure, and a review of the design for possible design changes.

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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

Any operation or condition prohibited by the plant's Technical Specifications.

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

At 1601 on July 20, 1992, Comanche Peak Steam Electric Station (CPSES) Unit 1 was manually tripped from 100 percent power. The details of this event are documented in Licensee Event Report (LER) 92-019.

At 1900 on July 20, 1992, CPSES Unit 1 was in Mode 3, Hot Standby, with the Reactor Coolant System (RCS)(EII:(AB)) at a temperature of 556 degrees Fahrenheit (F) and pressure of 2241 pounds per square inch-gage. The Motor Driven Auxiliary Feedwater Pumps (MDAFWP) (EIIS:(P)(BA)), 1-01 and 1-02, were in operation, maintaining Steam Generator (SG) (EIIS:(SG)(SB)) water levels.

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 1900 on July 20, 1992, the Reactor Operator (RO) (utility, licensed) noticed that the output of the MDAFWP 1-02 was approximately 100 gallons per minute (gpm) higher than the combined flow rate to the steam generators, SG-03 and 04, it was feeding. The RO anticipated that the flow mismatch was the result of a false indication from the MDAFWP 1-02 discharge flow transmitter (EIIS:(FT)(BA)). At 1930 the RO dispatched an Instrumentation and Control (I&C) technician (utility, nonlicensed) to fill and vent the flow transmitter. The fill and vent was successfully completed, however the flow mismatch was not affected. The I&C technician was then dispatched to troubleshoot the flow transmitters monitoring flow to SG-03 and 04. These flow transmitters were found to be operating correctly.

At approximately 1900 on July 21, 1992, with the flow transmitters verified to be operating correctly, the RO began to look elsewhere for the cause of the flow mismatch. The RO dispatched an Auxiliary Operator (AO) (utility, nonlicensed) to check on the status of the MDAFWP 1-02 test line isolation valve (1AF-0055) (EIIS:(ISV)(BA)). At 1959 on July 21, 1992, the AO found 1AF-0055 to be three-eighths of a turn open with the lock seal still intact. The AO closed 1AF-0055. As a result, the flow mismatch between the output of MDAFWP 1-02 and the flow to SG-03 and 04, was eliminated.

1AF-0055 was last opened during an Auxiliary Feedwater (AFW) System (EIIS:(BA)) operability test on July 2, 1992. The valve was opened, locked closed, and independently verified closed at the conclusion of the test. There have been no work clearances or tests performed between July 2, 1992, and July 21, 1992, that would require 1AF-0055 to be operated.

Engineering judgement determined that approximately 100 gpm was going through 1AF-0055, and that MDAFWP 1-02 was therefore inoperable. With the position of 1AF-0055 unchanged since July 2, 1992, MDAFWP 1-02 was inoperable from July 2, 1992, to July 21, 1992. CPSES Unit 1 Technical Specifications (TS) require three independent steam generator auxiliary feedwater pumps and associated flow paths, to be operable in Modes 1, 2, and 3. CPSES Unit 1 was in Modes 1, 2, and 3 from July 2, 1992, to July 21, 1992. As a result, the 72 hour TS Action Statement for one auxiliary feedwater pump or associated flow path inoperable, was exceeded.

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

At 1900 on July 20, 1992, the RO noticed the flow mismatch. The RO then had I&C verify the affected flow transmitters. The transmitters were verified to be operating correctly. At 1959 on July 21, 1992, an AO discovered 1AF-0055 open three-eighths of a turn. The AO closed 1AF-0055 and the mismatch was eliminated.

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II. COMPONENT OR SYSTEM FAILURES

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

Not applicable - there were no component failures associated with this event.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Not applicable - there were no component failures associated with this event.

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 that affected this event.

D. FAILED COMPONENT INFORMATION

Not applicable - there were no component failures associated with this event.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

Not applicable - there were no safety system actuations associated with this event.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

MDAFWP 1-02 was inoperable from July 2, 1992, to July 21, 1992 (19 days).

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The AFW System is designed to provide a supply of high-pressure feedwater to the secondary side of the steam generators for reactor coolant heat removal following a loss of normal feedwater. After a loss of the main feedwater system (EIIS:(SJ)), either the two motor-driven auxiliary feedwater pumps together, the turbine-driven pump alone, or any combination are capable of providing sufficient flow to the steam generators to allow the plant to be taken to a safe shutdown condition.

MDAFWP 1-01 and the Turbine-Driven Auxiliary Feedwater Pump (TDAFW) (EIIS:(P)(BA)) were operable during the time that MDAFWP 1-02 was inoperable; therefore the AFW System could perform its intended safety function. At no time during this period did an actual condition exist that threatened the health or safety of the public.

IV. CAUSE OF THE EVENT

ROOT CAUSE

The root cause of this event was a problem not anticipated with the design of the operating apparatus of 1AF-0055. This valve is operated by a floor stand with a 24 inch diameter handwheel. There are three drive shafts connecting four gear boxes from the handwheel to the valve. The handwheel must be turned counterclockwise to close the valve. This is clearly labeled. Due to the diameter of the handwheel and the gear reduction, the use of a leverage device is not permitted. With this design it is difficult to determine if the valve is closed. The "feel" that the valve is closed is not reliable due to the numerous mechanical devices. The AFW System operability test procedure required 1AF-0055 to be closed after the MDAFWP was shut down in order to reduce differential pressure across the valve. As a result, 1AF-0055 could not be verified closed other than by "feel". Other similar valves provide positive indication to the Operator, by "feel".

V. CORRECTIVE ACTIONS

A. CORRECTIVE ACTIONS TO PREVENT RECURRENCE

ROOT CAUSE

The root cause of this event was a problem not anticipated with the design of the operating apparatus of 1AF-0055.

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CORRECTIVE ACTION

On July 30, 1992, the AFW System operability test was performed to determine if 1AF-0055 could be shut with MDAFWP 1-02 in operation. The valve was shut against the differential pressure with a minimum of additional effort. When 1AF-0055 was shut, the RO in the Control Room verified the flow through the test line had dropped to zero. The valve was then locked in that position.

A change to the AFW System operability test procedure has been implemented to reflect this new method to positively verify 1AF-0055 closure.

The design of the operating apparatus of 1AF-0055 will be analyzed and a design change implemented as required.

VI. PREVIOUS SIMILAR EVENTS

LER 91-10, "Unit 1 Operated Outside Technical Specifications Due To Auxiliary Feedwater System Test Line Isolation Valve Not Fully Closed", documents an event similar to the event described in this LER (92-20). The root cause of LER 91-10 could not be determined; however, three possibilities were listed which involved: 1) Failure to fully close 1AF-0055 when restoring from a test; 2) The valve was disturbed after it was closed; and 3) Vibration opened the valve. The corrective actions for LER 91-10 involved: 1) Issuing a Lesson's Learned memo to Shift Operations personnel instructing the operators to ensure that 1AF-0055 was closed; 2) Locking the valve closed; and 3) An engineering analysis of the vibrations on 1AF-0055.

This LER (92-20) is similar to LER 91-10 in that 1AF-0055 was not completely closed after a test. In LER 91-10 the corrective action was addressed toward the Operations Shift personnel. In this LER (92-20) it has been determined that the design of the operating apparatus of 1AF-0055 did not provide reliable indication to the Operator, by "feel", that the valve was closed. The corrective actions taken in this LER (92-20) should prevent recurrence.

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.

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VII. ADDITIONAL INFORMATION

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