TU ELECTRIC

William J. Cahill, Jr.
Executive Vice President

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Ref. # 50.73 (a)(2)(iv)

April 16, 1991

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE
LICENSEE EVENT REPORT 91-008-00

Gentlemen:

Enclosed is Licensee Event Report 91-008-00 for Comanche Peak Steam Electric Station Unit 1, "Reactor Trip Caused by Personnel Error During Testing."

Sincerely,

W. J. Cahill Jr.

JAA/bm

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

IE22 1/1

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)

REACTOR TRIP CAUSED BY PERSONNEL ERROR DURING TESTING

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 3	1 7	9 1	9 1	- 0 0 8 -	0 0	0 4	1 6	9 1	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

Operating Mode (9): 1

Power Level (10): 0 9 7

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 Area Code	
T. A. HOPE	COMPLIANCE SUPERVISOR	8 1 7	8 9 7 - 6 3 7 0

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	

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 March 17, 1991, Comanche Peak Steam Electric Station Unit 1 was in Mode 1, Power Operation, with reactor power at 97 percent. An Auxiliary Operator and trainee were performing a Solid State Protection System, Train A, actuation logic test. During the test, the trainee depressed the shunt trip button for the Train B reactor trip breaker, instead of the intended Train A trip button, causing a reactor trip. Root causes were identified as failure to verify the correct equipment and the Auxiliary Operator allowing his job as a trainer to interfere with his testing duties. Corrective actions include self verification training, enhanced breaker cabinet labels, monitoring operators to ensure self verification techniques are utilized, and additional guidance for trainers on handling trainees.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
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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

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

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On March 17, 1991, just prior to the event, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, with reactor power at 97 percent. Solid State Protection System (SSPS) (EIIS:(JC)), Train A, actuation logic test was in progress.

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 to the event.

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

At 1129 on March 17, 1991, an Auxiliary Operator (AO) (utility, non-licensed) and a trainee (utility, non-licensed) were performing a SSPS, Train A, actuation logic test. Three-way communication was established between the AO and the trainee at the Reactor Trip Switchgear cabinet (EIIS:(JC)(CAB)), and the Reactor Operator (RO) (utility, licensed) in the Control Room. The AO was checking off each step as it was completed and was at the side of the trainee at all times. The AO checked each step and gave concurrence prior to the trainee performing the step. The step in question required pushing the shunt trip for Reactor Trip Breaker A (EIIS:(JC)(BKR)). The expected results would be for Reactor Trip Breaker A to open and the control rods to remain energized because Reactor Trip Bypass Breaker A (EIIS:(JC)(BKR)) would be closed. Instead, the AO and trainee failed to verify that they were in the correct cabinet. They were actually in the Train B cabinet. The shunt trip was pushed on Reactor Trip Breaker B (EIIS:(JC)(BKR)) instead of the intended Breaker A, causing

Breaker B to open and all control rods to drop.

Control Room personnel responded in accordance with emergency operating procedures. Plant systems responded as expected. The plant was stabilized in Mode 3, Hot Standby, at 1210. At 1317 the NRC was notified of the event via the Emergency Notification System in accordance with 10CFR50.72.

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

The reactor trip was annunciated by numerous alarms in the Control Room. The immediate cause of the trip was reported by the AO.

II. COMPONENT OR SYSTEM FAILURES

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

No failed components contributed to this event.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

No failed components contributed to this event.

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

No failed components contributed to this event.

D. FAILED COMPONENT INFORMATION

No failed components contributed to this event.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The Reactor Protection System (EIIS:(JC)) and Auxiliary Feedwater System (EIIS:(BA)) actuated during the event; all associated components within these systems functioned as designed.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

No safety system trains were inoperable as a result of this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The reactor trip was the result of a Reactor Trip Breaker shunt trip caused by personnel error during testing. The event of March 17, 1991, occurred at 97 percent reactor power, and all protective functions responded as designed. The event is completely bounded by the FSAR accident analysis which assumes an initial power level of 102 percent and conservative assumptions which reduce the capability of safety systems to mitigate the consequences of the transient. The event of March 17, 1991, did not adversely affect the safe operation of CPSES Unit 1 or the health and safety of the public.

IV. CAUSE OF THE EVENT

A. ROOT CAUSE

1. The trainer and trainee failed to verify the correct equipment as shown on the color coded label on the breaker cabinets.

2. The AO allowed his job as a trainer to affect his job as an Operator. The AO was distracted from concentrating on the test by imparting experience to the trainee during the evolution.

GENERIC CONSIDERATIONS

of self verification has been a root cause in other events.

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

V. CORRECTIVE ACTIONS

CORRECTIVE ACTIONS TO PREVENT RECURRENCE

A. ROOT CAUSE

1. The trainer and trainee failed to verify the correct equipment as shown on the color coded label on the breaker cabinets.

CORRECTIVE ACTION

All Operators have been briefed on self verification techniques again by Shift Operations management.

The following corrective actions have been implemented to enforce self verification:

- requiring the trainer to ensure that the trainee performs self verification throughout the task,
- making the trainee responsible to perform self verification at all times,
- requiring the trainer to ensure that the Control Room is aware of training evolutions in the plant,
- enhancing breaker cabinet labeling by installing larger color coded labels on the outside and adding larger labels on the inside of the cabinet next to the shunt trip button, and
- monitoring operators on a regular basis to ensure that self verification techniques are being utilized.

2. The AO allowed his job as a trainer to affect his job as an Operator.

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CORRECTIVE ACTION

Discussions have been held with Operations Supervisors on ways to improve the trainer's control of trainees during evolutions. As a result of these discussions, corrective actions have been implemented, including:

- requiring the trainer to ensure that the trainee performs self verification throughout the task, and
- training the trainers on techniques to maintain control of an evolution while instructing a trainee.

B. GENERIC CONSIDERATIONS

Lack of self verification has been a root cause in other events.

CORRECTIVE ACTION

A task team, formed to evaluate several events including a reactor trip reported in Licensee Event Report (LER) 91-004, concluded that inadequate self verification has been a continuing problem in recent months. As a result of the "Reactor Trip and Plant Incident Assessment" task team report, Operations management has mandated self verification training and followup evaluation by shift management. Inadequate self verification or a lack of "attention to detail" by operators and technicians is being monitored on a daily basis by Operations management to evaluate the effectiveness of the corrective actions. A report on self verification root cause trending will be performed in the future to verify that the corrective actions outlined in this LER have been effective.

VI. PREVIOUS SIMILAR EVENTS

LER 91-004 also involved a reactor trip caused by an AO and trainee. The initial conditions were different. This LER occurred during testing, while following an approved procedure and while in contact with the Control Room; LER 91-004 occurred during informal training, using no approved procedure and not informing the Control Room of the

training. As a result of the "Reactor Trip and Plant Incident Assessment" task team report, corrective actions have been implemented to reduce personnel error. The corrective actions, the same as described in this LER, are long term solutions that have not had sufficient time to take effect. As a result, the corrective actions identified in the task team report, given sufficient time to take effect, could have prevented this event.

LER 90-007 involved an ESF actuation that was attributed to lack of adequate self verification. This event involved lack of self verification of a procedural step instead of a piece of equipment; however, it is an example of a lack of "attention to detail" that is actively being addressed by Operations management (see Generic Considerations above).

VII. ADDITIONAL INFORMATION

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