TU ELECTRIC

Log # TXX-90219
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Ref. # 50.73 (a)(2)(iv)

July 6, 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 90-018-00

Gentlemen:

Enclosed is Licensee Event Report 90-018-00 for Comanche Peak Steam Electric Station Unit 1, "Inadvertent Automatic Start of Auxiliary Feedwater Pump Due to Personnel Error."

Sincerely,

William J. Cahill

William J. Cahill, Jr.

KWV/daj

Enclosure

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

IE22
1/1

P.O. Box 1002 Glen Rose, Texas 76043-1002

Enclosure to TXX-90219

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)

Facility Name (1)	Docket Number (2)	Page (3)
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Title (4)
INADVERTENT AUTOMATIC START OF AUXILIARY FEEDWATER PUMP DUE TO PERSONNEL ERROR

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

Operating Mode (9): 5
Power Level (10): 0 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 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

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 June 13, 1990, Comanche Peak Steam Electric Station Unit 1 was in Mode 5; two steam generators had been drained to accommodate chemistry control activities. During performance of Train A Safeguards slave relay testing activities, Operations personnel discovered that the Solid State Protection System was not aligned to support test performance. When the operator attempted to establish the required alignment, the Train A Motor Driven Auxiliary Feedwater Pump started as a result of Lo Lo level in the drained steam generators. Causes of the event are personnel error and procedural inadequacies. Corrective actions include event review by Operations personnel and procedural enhancements.

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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 June 13, 1990, at 0316 Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 5, Cold Shutdown, with the Reactor Coolant System (RCS) (EIIS:(AB)) at a temperature of approximately 145 °F and pressure of 350 psig.

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

Steam Generators (EIIS:(SG)(SB)) 1 and 3 had been drained previously to accommodate chemistry control activities unrelated to the event. To accommodate plant conditions the Solid State Protection System (SSPS) (EIIS:(JE)) had been placed in the "Mode 5/6" lineup by placing the Mode Selector Switch (S602) (EIIS:(JE)(33)) in the Test position. This removes the 118 volts AC normally available to the Engineered Safety Features (ESF) actuation relays (EIIS:(JE)(RLY)). The "Mode 5/6 - Normal" switch (S604) (EIIS:(JE)(33)) was in the "Mode 5/6" position; in this position 118 volts AC is available to the actuation relays associated with the ESF functions required in Modes 5 and 6.

C. EVENT CLASSIFICATION

An event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature.

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

On June 13, 1990, just prior to 0316, Control Room personnel were performing Train A Safeguards slave relay actuation testing. Upon relay test actuation the expected response was not observed. Personnel determined that S602 was not positioned to allow test performance. The system operating procedure for the SSPS was consulted and the Reactor Operator (utility, licensed) performed the procedure section for establishing the lineup required to allow performance of the test in progress. In accordance with

procedural guidance the Reactor Operator checked to ensure that required blocks were in place, and that the annunciator windows (EIIS:(ANN)(IB)) and Trip Status Light Boxes (TSLB) (EIIS:(AN)(IB)) reflected actual plant conditions. When the Reactor Operator placed S602 in the "Operate" position, the Train A Motor Driven Auxiliary Feedwater (MDAFW) Pump (EIIS:(BA)(P)) started on an ESF actuation signal as a result of the Lo-Lo Level in Steam Generators 1 and 3. The Reactor Operator immediately recognized the cause of the pump start and secured the pump by placing the control switch (EIIS:(BA)(33)) in the "Pull-Out" position. After a shift group discussion of the event, testing was successfully completed with no further problems.

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

The system operating procedure for the SSPS instructs the operator to consult a procedure attachment "to determine which ESF functions have actuation signals based on TSLB status." The attachment contains information to advise the operator that the Steam Generator Lo-Lo level reactor trip and Auxiliary Feedwater Pump start cannot be blocked. During the group discussion following the MDAFW pump start, the Unit Supervisor (utility, licensed) and Reactor Operator realized that the referenced attachment had not been adequately reviewed prior to manipulation of S602.

During the investigation to determine causes and contributing factors leading to the MDAFW Pump Start, procedural weaknesses were identified which contributed to establishing the conditions which led to the event.

II. COMPONENT OR SYSTEM FAILURES

A. FAILED COMPONENT INFORMATION

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

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

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

C. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

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

D. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURES OF COMPONENTS WITH MULTIPLE FUNCTIONS

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

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The Train A MDAFW Pump auto-started on Lo Lo steam generator level, and the Condensate Storage Tank isolation valve (EIIS:(KA)(V)) (to Condenser) closed; however, no Auxiliary Feedwater flow to the steam generators occurred because the Auxiliary Feedwater isolation valves were tagged closed. The Turbine Driven Auxiliary Feedwater Pump is designed to start on Lo Lo level in any 2 steam generators; however, it was tagged out-of-service. The blowdown valves, sample valves (EIIS:(KN)(V)), and feedwater split flow bypass valves (EIIS:(SJ)(FCV)) which receive a close signal were already closed during the reactor shutdown.

B. DURATION OF SAFETY SYSTEM INOPERABILITY

Not applicable - there were no safety systems required for existing plant conditions which were rendered inoperable by this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The Auxiliary Feedwater System is required to be operable in Modes 1, 2 and 3 to ensure that the Reactor Coolant System can be cooled from normal operating temperature to less than 350 degrees F. The worst consequences of an inadvertent AFW pump start result from mixing of the relatively cooler AFW with Main Feedwater, a condition leading to an increase in heat removal by the secondary system as described in the accident analysis discussed in Chapter 15 of the Final Safety Analysis Report (FSAR).

The alignment of switches S602 and S604 prior to the event is procedurally allowed only in plant operating modes 5 and 6. There is, therefore, reasonable assurance that the conditions which led to the event would not have occurred at a more severe set of initial conditions, and it is concluded that the consequences of the inadvertent actuation are completely enveloped by the analysis presented in Chapter 15 of the FSAR.

The actuation of the AFW system demonstrates that the system would have performed as designed if the actuation had been the result of one of the plant conditions for which the system is required to respond. It is concluded that 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 is a less than adequate personnel performance. Operations personnel failed to adequately utilize the procedure attachment containing the information which would have prompted the actions required to prevent the ESF actuation.

CONTRIBUTING FACTORS

Less than adequate procedural guidance is considered to be a contributing factor to the event. While not directly responsible for the event, the following procedural omissions contributed to establishing the conditions which led to the inadvertent ESF actuation:

1. The system operating procedure used to drain Steam Generators 1 and 3 noted the ESF actuation signal associated with Lo Lo Steam Generator Level, but did not provide specific instruction to place the MDAFW pump controls in "Pull-Out."

2. The slave relay actuation test procedure in progress just prior to the event did not contain sufficient prerequisite information to ensure successful test completion. The procedure did not require the operator to verify the correct position of switches S602 and S604 prior to test performance.

V. CORRECTIVE ACTIONS

A. IMMEDIATE ACTIONS

The Reactor Operator immediately secured the Train A MDAFW Pump by placing the control switch in the "Pull-Out" position.

B. ACTIONS TO PREVENT RECURRENCE

Cause: Less than adequate personnel performance

Corrective Action: A "Lessons Learned" memorandum describing the event and its causes has been developed for review by all Control Room operating personnel. The Plant Incident Report (PIR) addressing this event has been placed in the PIR Log in the Control Room for review by all Control Room operating personnel.

Contributing Factor: Procedural inadequacy

Corrective Action: The system operating procedure used to drain the steam generators has been changed to ensure that MDAFW pump controls are properly positioned to prevent an auto-start on Lo Lo Steam Generator Level.

The test procedure in use at the time of the event was changed to provide additional prerequisite information. A review of other slave relay test procedures was performed to identify similar procedural inadequacies. Those procedures identified as having similar weaknesses were enhanced to ensure that the prerequisites place the system in a configuration accommodating test performance.

During the investigation of this event, it was recognized that the system operating procedure for the SSPS was somewhat confusing for operators to use. A Human Factors review was performed to identify potential problems with the procedure and to develop suggestions for improvements. Although not considered to be a direct contributor to the inadvertent MDAFW pump start, the procedure has been revised to improve ease of use for operating personnel.

VI. PREVIOUS SIMILAR EVENTS

LER 90-007-00 described an ESF actuation resulting from an inadequate review of impact on the plant prior to the de-energization of a radiation monitor. However, the details of the event described in LER 90-007 and the resultant corrective actions were sufficiently different from those of this LER to conclude that the previous corrective actions could not be expected to have prevented the inadvertent start of the MDAFW pump described in this report.

VII. ADDITIONAL INFORMATION

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