TU ELECTRIC

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

February 26, 1993

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
REACTOR PROTECTION SYSTEM ACTUATION
LICENSEE EVENT REPORT 93-001-01

REF: TU Electric letter logged TXX-93100 from W. J. Cahill, Jr. to USNRC dated February 16, 1993

Gentlemen:

Enclosed is Licensee Event Report 93-001-01 for Comanche Peak Steam Electric Station Unit 1, "Reactor Trip Caused by Personnel Error during Solid State Protection System Testing."

The subject Licensee Event Report has been revised to clarify the corrective actions specified in the referenced letter. This matter was discussed with Mr. T. Reis of your staff.

Sincerely,

William J. Cahill, Jr.

By: J. J. Kelley

J. J. Kelley, Jr.
Vice President of Nuclear Operations

020114

OB/tg
Enclosure

c - Mr. J. L. Milhoan, Region IV
Mr. L. A. Yandell, Region IV
Resident Inspectors, CPSES (2)

IE22 1/1

Enclosure to TXX-93113

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 SOLID STATE PROTECTION SYSTEM 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 Number(s)
01	18	93	93	- 001 -	01	02	26	93	N/A	05000
									N/A	05000

Operating Mode (9): 1

Power Level (10): 100

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	Area Code	Telephone Number
D. J. REIMER, MANAGER, SYSTEM ENGINEERING	817	897-5584

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)		Expected Submission Date (15)	Month	Day	Year
☐ Yes (If yes, complete Expected Submission Date)	☒ No				

Abstract (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

At 1144 on January 18, 1993, while restoring the Solid State Protection System (SSPS) from a test lineup, the Reactor Operator used an incorrect instruction. As a result, trip blocks were not in place for switch manipulation and a reactor trip occurred while restoring SSPS to a normal lineup.

Root cause of the event was personnel error. Contributing factors were that a test procedure reference was not specific which led the RO to an incorrect instruction and that senior supervision monitoring during the restoration was less than adequate. Corrective actions included event review, training and procedure revision.

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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 (RPS)(EIIS:(JC)).

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On January 18, 1993, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, with reactor power at 100 percent.

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 1130 on January 18, 1993, a Train A Solid State Protection System (SSPS)(EIIS:(JG)) logic test had just been completed and the Reactor Operator (RO)(utility, licensed) was proceeding to return SSPS to a Normal lineup. The Shift Technical Advisor/Unit Supervisor (STA/US)(utility, licensed), a Senior Reactor Operator, was supervising the RO during testing as required by plant management for activities identified as an "infrequent evolution".

The test procedure referenced the Standard Operating Procedure (SOP) for SSPS for instructions on how to restore SSPS to a Normal lineup. The test procedure did not reference a specific section of the SOP and the Reactor Operator went to an attachment titled, "Control Switch Lineup Sheet - Normal Lineup", which he had correctly used earlier in the test procedure to verify the initial SSPS switch lineup prior to starting the actual test. In this case, however, the RO should have gone to the body of the procedure to the section titled, "Disabled Lineup to Normal Lineup OR

Disabled Lineup to Mode 5/6 Lineup." The attachments of the SOP are verification sheets only and were not to be used for switch manipulation. As a result of using the verification sheet, the required trip block was not in place when the RO took the INPUT ERROR INHIBIT switch(EIIS:(33)(JG)) to NORMAL. Both the RO and STA/US had incorrectly assumed that since the Mode Selector switch on the SSPS Output Relay Test Panel was in TEST that the required blocks were in place. The STA/US had also started reviewing test data at the conclusion of the test and was not paying as close attention to the RO during restoration as he did during the actual test. The STA/US believed that his responsibilities were primarily to monitor the actual test and that the RO would not require monitoring to restore SSPS to the Normal lineup. At 1144, the RO took the INPUT ERROR INHIBIT switch to NORMAL and a reactor trip occurred.

Following the trip, Control Room personnel responded in accordance with emergency operating procedures. Plant systems responded as expected. The plant was stabilized in Mode 3, Hot Standby. An event or condition that results in an automatic actuation of any ESF, including the RPS, is reportable within 4 hours under 10CFR50.72(b)(2)(ii). At 1250 on January 18, 1993, the Nuclear Regulatory Commission Operations Center was notified of the event via the Emergency Notification System.

The investigation following the trip revealed that the previous revision of the test procedure had referenced the specific section in the SOP for restoring SSPS to a Normal lineup. The procedure was revised and the reference removed because of problems encountered when procedure sections changed numbers. When a procedure section number was changed, it required that all other procedures referencing that section would also have to change. By removing the specific section number from the referencing procedures this problem was solved; however, the alternative was inadequate in that it led to confusion in this instance.

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

The reactor trip was annunciated by numerous alarms in the Control Room. The cause of the trip was immediately known.

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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

A Lo-Lo Steam Generator level signal was generated, resulting in an ESF actuation; actuating the Auxiliary Feedwater System (AFW)(EIIS:(BA)). Associated components within these systems functioned as designed. Train B AFW actuated automatically, Train A did not autostart as a result of the configuration of Train A SSPS (the Mode Selector switch was still in TEST). This condition was immediately recognized and Train A AFW was manually started at 1147.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Train A SSPS was inoperable during the logic test. After the trip, Train A AFW did not autostart because Train A SSPS had not yet been restored. This was quickly recognized and Train A AFW was manually started within 3 minutes following the trip and Steam Generator levels were restored.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The reactor trip was the result of personnel error and was not required to mitigate an actual event. This event is best described in Section 15.2.3 of the CPSES Final Safety Analysis Report (FSAR). The analysis uses conservative assumptions to demonstrate the Departure from Nucleate Boiling Ratio will never decrease below the limiting value of 1.30 during the event. The event of January 18, 1993, occurred at 100 percent reactor power, and all protective functions responded as required. The event is completely bounded by the FSAR accident analysis. The event of January 18, 1993, 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

Root cause of the event was personnel error. The RO and STA/US were not sufficiently familiar with the SSPS circuitry in that they incorrectly assumed that sufficient trip blocks were in place so that the INPUT ERROR INHIBIT switch could be taken to NORMAL without a reactor trip occurring.

CONTRIBUTING FACTORS

1. The test procedure did not make a specific enough reference to the SOP for SSPS restoration. This led the RO to an incorrect set of instructions which led to switch manipulation without the required trip blocks being established.

2. The STA/US did not monitor the restoration of the test lineup as closely as he had monitored the actual test. The STA/US was reviewing the test results and did not pay close enough attention to catch the mistake.

GENERIC CONSIDERATIONS

The familiarity with the specifics of the SSPS circuitry is not consistent among operators.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92
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Text (If more space is required, use additional NRC Form 366A's) (17)

V. CORRECTIVE ACTIONS TO PREVENT RECURRENCE

ROOT CAUSE

The RO and STA/US were not sufficiently familiar with the SSPS circuitry.

CORRECTIVE ACTION

The RO and STA/US have received personal counseling. A "Lessons Learned" was issued for shift personnel on the details of the event. Formal training is being revised to include aspects from this event and to include more detailed objectives and lessons for operator understanding of the SSPS circuitry.

CONTRIBUTING FACTOR - 1

The test procedure did not make a specific enough reference to the SOP for SSPS restoration.

CORRECTIVE ACTION

The test procedure was enhanced to provide explicit procedure transitions to the SOP for the SSPS.

A note was placed with all the verification attachments in the SOP stating that the attachments are to be used for verification only and not for actual switch manipulation.

CONTRIBUTING FACTOR - 2

The STA/US did not monitor the restoration of the test lineup as closely as he had monitored the actual test.

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CORRECTIVE ACTION

Lessons Learned and formal training will stress that the requirements in place for "infrequent evolutions" include active monitoring until the evolution is complete and equipment is restored to a normal steady-state lineup. The SOP for SSPS has been redefined as an "infrequent evolution" and will now require the same monitoring requirements as SSPS testing (e.g., evolutions performed by experienced personnel, prebriefing, direct supervisory involvement).

GENERIC CONSIDERATIONS

The familiarity with the specifics of the SSPS circuitry is not consistent among operators.

CORRECTIVE ACTION

A "Lessons Learned" was issued for shift personnel on the details of the event. Formal training is being revised to include aspects from this event and to include more detailed objectives and lessons for operator understanding of the SSPS circuitry.

VI. PREVIOUS SIMILAR EVENTS

In Licensee Event Report (LER) 92-005 a RO was performing slave relay testing. The RO misinterpreted a procedure step and inadvertently unblocked the actuation signal which had been blocked in a previous procedure step. Upon performance of the next procedure step, all four feedwater split flow bypass valves closed. The root cause of this event was also personnel error; however, several of the lessons learned from LER 92-005 have been successfully incorporated. The extenuating circumstances in this event was the confusion with the recently revised test procedure reference and that the SRO (STA/US) monitoring the evolution remained vigilant only until the actual test was complete and did not monitor the restoration as closely.

VII. ADDITIONAL INFORMATION

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