TU ELECTRIC

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January 24, 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
CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS
LICENSEE EVENT REPORT 91-030-01

Gentlemen:

Enclosed is Licensee Event Report 91-030-01 for Comanche Peak Steam Electric Station Unit 1, "Personnel Error Leading to Mispositioned Residual Heat Removal System Crosstie Valves".

Sincerely,

William J. Cahill, Jr.

William J. Cahill, Jr.

NSH/tg

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (2)

Enclosure to TXX-92056

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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Title (4): PERSONNEL ERROR LEADING TO MISPOSITIONED RESIDUAL HEAT REMOVAL SYSTEM CROSSTIE VALVES

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
1 2	0 4	9 1	9 1	- 0 3 0 -	0 1	0 1	2 4	9 2	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

Operating Mode (9): 3

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 Area Code	
D.E. BUSCHBAUM COMPLIANCE SUPERVISOR	8 1 7	8 9 7 - 5 8 5 1

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)		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)

On December 4, 1991, Comanche Peak Steam Electric Station Unit 1 entered Mode 3 with two mispositioned valves in the Emergency Core Cooling System. The event is considered to be a failure to satisfy a Limiting Condition for Operation and a surveillance requirement of the plant's Technical Specification. The cause of the event has been determined to be personnel error leading to the failure to properly position the crosstie valves in the Residual Heat Removal System following filling of a portion of the system. Corrective actions include training and procedure enhancement.

Enclosure to TXX-92056

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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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

On December 4, 1991 (Event Date), at 1333 CST, Comanche Peak Steam Electric Station Unit 1 was declared to be in Mode 3, Hot Standby.

On December 6, 1991 (Discovery Date), at 1615 CST, Unit 1 was still in Mode 3 in preparation for a plant startup.

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

On December 4, 1991, following the first refueling outage, activities were in progress to bring the plant to Hot Standby. Various valves in the Emergency Core Cooling System (ECCS) (EIIS:(BP)) associated with the Residual Heat Removal (RHR) system (EIIS:(BP)) were placed in the required Mode 3 alignment in accordance with the integrated plant operating procedures. Train A of the RHR system had previously been placed in Standby Readiness, and Train B was operating in the shutdown cooling mode. RHR crosstie valve 1 (refer to Figure 1) (EIIS:(V)(BP)) was open and crosstie valve 2 was closed in accordance with the alignment specified in the system operating procedure.

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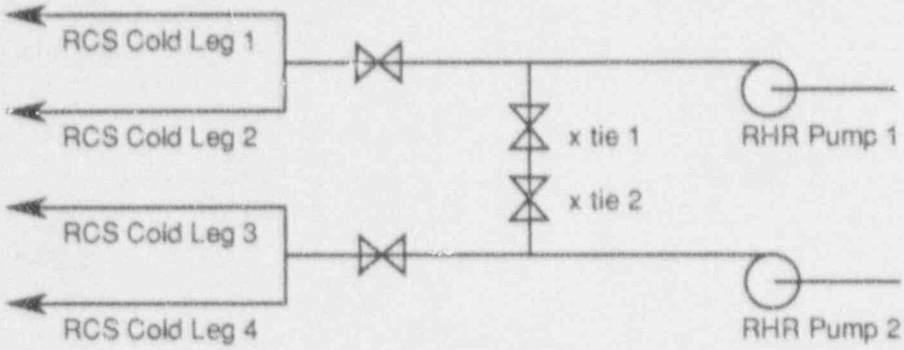

Figure 1

In preparation for testing of several check valves, Train B of the RHR system was secured. Difficulties encountered during check valve testing necessitated realignment of the system to allow the discharge header to be filled. During this activity, crosstie valve 1 was closed. Following completion of check valve testing, the RHR pumps and system were vented to satisfy the related surveillance requirement.

The Reactor Operator (utility, licensed) was directed by the Unit Supervisor (utility, licensed) to place the RHR system in standby readiness in accordance with the system operating procedure. While performing the alignment, the Reactor Operator failed to complete all steps necessary to place the RHR system in the required alignment, and inadvertently left the RHR crosstie valves closed. At 1333 the plant was declared to be in Mode 3.

Technical Specification 4.5.2b requires that each valve in the ECCS "flow path that is not locked, sealed, or otherwise secured in position, be verified in its correct position" at least once per 31 days when the plant is in modes 1, 2, or 3. Technical Specification 4.0.4 states that "Entry into an [operational mode] or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified." Entry into mode 3 with the RHR crosstie valves closed represents a failure to satisfy the requirement of Technical Specification 4.0.4 as set forth in Specification 4.5.2b. Technical Specification 3.0.4 prohibits entry into an operational mode when the conditions for the Limiting Conditions for Operation are not met. Entry into mode 3 with the RHR crosstie valves closed represents a failure to satisfy the requirement of Technical Specification 3.0.4.

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

While reviewing plant systems configurations on the Emergency Response Facility (ERF) computer, an engineer (utility, non-licensed) in the Instrument and Control (I&C) group observed that the RHR crosstie valves were not in the position expected with the plant in Mode 3. The I&C engineer contacted a member of the Independent Safety Engineering Group (ISEG) to raise the question of proper valve position. After review of the related operating procedures to confirm the correct valve position, the ISEG engineer (utility, non-licensed) contacted the Control Room.

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 component failures associated with 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

The RHR crosstie valves remained mispositioned for approximately 53 hours and 7 minutes. This condition did not result in the inability of safety systems or components to perform their intended functions.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

Operability of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a loss of coolant accident (LOCA), assuming the loss of one subsystem through any single failure. The ECCS analysis assumes low head safety injection into all four cold legs of the Reactor Coolant System (RCS). For the limiting cold leg break location, safety injection flow into the ruptured loop is assumed to spill to the containment with flow to the remaining three cold legs.

An engineering evaluation was performed to determine if plant response to a potential LOCA while in Mode 3 would be adversely affected with the RHR crosstie valves closed. The evaluation uses conservative assumptions and identifies the worst case break site and location for the Mode 3 LOCA. The evaluation considers Mode 3 operation at pressure and temperature conditions with blocked accumulators and the requirement for manual initiation of safety injection. The evaluation also considers Mode 3 operation at higher pressure and temperature conditions with accumulators and automatic safety injection available. In each case it is concluded that the failure to open the RHR crosstie valves on December 4, 1991 did not adversely affect the ability of the plant to recover from a Mode 3 LOCA.

Entry into Mode 1, Power Operation, with the RHR crosstie valves closed is considered unlikely; the misalignment would have been discovered prior to entry into Mode 2. The Train A RHR monthly verification surveillance was scheduled for performance on December 6, 1991, and the mispositioned crosstie valves would

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have been found as a result of that surveillance activity. Nonetheless, additional analysis was performed using the TU Electric Licensing methodology to determine the impact on cladding temperature following a large break LOCA at normal operating temperature and pressure. Analysis demonstrates that failure to open the RHR crosstie valves with the reactor at 25% of rated thermal power does not inhibit the ability to recover safely from the event. It is concluded that the failure to open the RHR crosstie valves prior to entry into Mode 3 on December 4 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 CAUSES

Root cause number 1: Personnel error resulted in the failure to place the RHR system in the required alignment prior to entry into Mode 3. The reactor operator did not complete the section of the RHR system operating procedure required to place the system in standby readiness.

Root cause number 2: Operating personnel did not document valve manipulations performed in support of testing activities. Operations Department administrative controls allow the manipulation of certain components in order to accomplish surveillance or other testing activities, corrective or preventive maintenance, or other operational activities as long as the position or state of the components is documented within the procedure controlling the activity or within the appropriate log.

B. CONTRIBUTING FACTORS

Contributing factor number 1: The reactor operator thought that the required valve and control switch lineups had been completed by a previous shift. The reactor operator reviewed the handswitch alignment and realized that several hand switches were out of position, but planned to realign the switches during restoration from testing activities.

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Contributing factor number 2: The surveillance program did not require an independent lineup to verify the proper position of RHR crosstie valves prior to entry into Mode 3. The system operating procedure was relied upon to place the system in the correct alignment within required TS time limits.

Contributing factor number 3: The reactor operator was involved in the various activities involving RHR system configuration changes while preparing for the infrequently performed activity of entry into Mode 3.

V. CORRECTIVE ACTIONS

A. IMMEDIATE

Upon notification and completion of research to determine the correct valve position, the Unit Supervisor directed that the RHR crosstie valves be opened and that the appropriate surveillance test be performed to verify the correct position of other valves in the system. A review of the surveillance database was performed to identify the potential for similar problems. Additional alignment verifications were performed to ensure that no other system configuration problems existed. No valve misalignments were found.

B. ACTIONS TO PREVENT RECURRENCE

Management expectations for operating personnel are being stressed to all crews using a variety of mechanisms such as shift orders, voice mail, training, and group discussions. The emphasis is on control board awareness, reliance on procedure implementation, and the need for ample and comprehensive log entries. These topics will be incorporated into requalification training for operating personnel.

A control switch alignment checklist has been developed to verify the correct position of various handswitches, controllers, etc., in the Containment Spray System, the RHR System, the Chemical and Volume Control System, and the Auxiliary Feedwater System. The checklist will be performed periodically during plant operation and prior to mode change during power ascension.

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Operations management will encourage scheduling of quiet periods prior to mode change on power ascension to allow Control Room personnel to review plant status and related documentation.

Various system operating procedures and operations testing procedures have been enhanced to minimize the potential for component misalignment. In addition, the control switch alignment checklist has been incorporated into the integrated operating procedure.

VI. PREVIOUS SIMILAR EVENTS

CPSES Licensee Event Reports (LER) 90-005, 90-010, 90-015, 90-024, 90-026, 90-034, 90-040, 90-044, 91-003, 91-007, 91-011, 91-017, and 91-028 describe previous events involving Technical Specification surveillance activities. The details of previously reported events are sufficiently different from the event described in LER 91-031 to conclude that previous corrective actions could not be expected to have prevented mispositioning the RHR crosstie valves.