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January 3, 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)
DUCKET NO. 50-445
CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS
LICENSEE EVENT REPORT 91-030-00

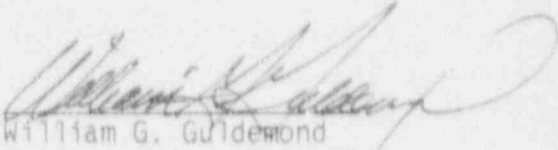
Gentlemen:

Enclosed is Licensee Event Report 91-030-00 for Comanche Peak Steam Electric Station Unit 1, "Personnel Error Leading to Mispositioned Residual Heat Removal System Cross Tie Valves".

Sincerely,

William J. Cahill, Jr.

By:


William G. Guidermond
Manager, Site Licensing

NH/tg

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

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NRC FORM 308		U.S. NUCLEAR REGULATORY COMMISSION		APPROVED OMB NO. 3150-0104 EXPIRES: 4/30/92	
LICENSEE EVENT REPORT (LER)				ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 60.0 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.	
Facility Name (1) COMANCHE PEAK - UNIT 1				Docket Number (2) 01510101014151	Page (3) 1 OF 016
Title (4) PERSONNEL ERROR LEADING TO MISPOSITIONED RESIDUAL HEAT REMOVAL SYSTEM CROSSTIE VALVES					
Event Date (5)		LER Number (6)		Report Date (7)	
Month	Day	Year	Year	Sequential Number	Revision Number
12	04	91	91	01310	0100110392
Other Facilities Involved (9)		Facility Name (8)			
Docket Number		N/A			
0151010101		N/A			
Operating Mode (10) 3 This report is submitted pursuant to the requirements of 10 CFR § (Check one or more of the following) (11):					
Power Level (12)		20.402(b)		20.405(c)	
01010		20.405(a)(1)(i)		50.73(a)(2)(iv)	
		20.405(a)(1)(ii)		50.73(a)(2)(v)	
		20.405(a)(1)(iii)		50.73(a)(2)(vi)	
		20.405(a)(1)(iv)		50.73(a)(2)(vii)(A)	
		20.405(a)(1)(v)		50.73(a)(2)(vii)(B)	
		20.405(a)(1)(vi)		50.73(a)(2)(viii)	
				50.73(a)(2)(ix)	
Licensee Contact For This LER (13) Name: D.E. BUSCHBAUM COMPLIANCE SUPERVISOR Area Code: 81117 Telephone Number: 819171-15181511					
Complete One Line For Each Component Failure Described in This Report (13)					
Cause	System	Component	Manufacturer	Reportable To NRC	Reportable To NRC
Supplemental Report Expected (14)					Expected Submission Date (15)
<input checked="" type="checkbox"/> Yes, (If yes, complete Expected Submission Date)					Month: 0 Day: 1 Year: 2492
<input type="checkbox"/> No					
Abstract (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)					
<p>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 will be determined based on the results of continuing evaluation.</p>					

NRC FORM 365A		U.S. NUCLEAR REGULATORY COMMISSION		APPROVED OMB NO. 3150-0104 EXPIRES: 4/30/92	
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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) (EIS:(B⁵)) associated with the Residual Heat Removal (RHR) system (EIS:(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) (EIS:(V)(BP)) was open and crosstie valve 2 was closed in accordance with the alignment specified in the system operating procedure.

NRC FORM 588A		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 588A's) (17)

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.

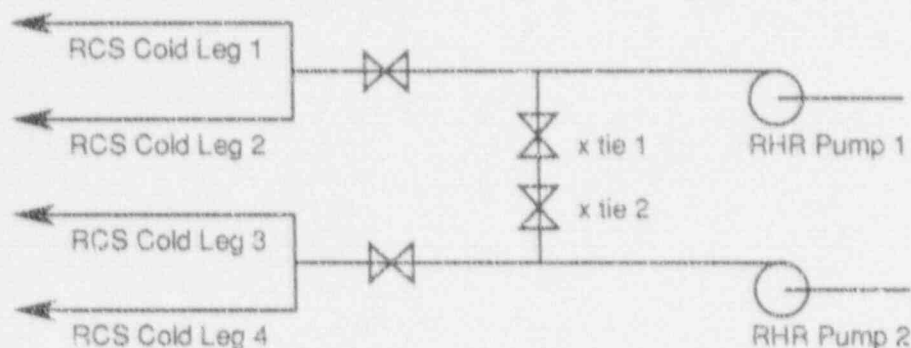


Figure 1

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 cross-tie 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 component failures associated with this event.

LICENSEE EVENT REPORT (LER) **TEXT CONTINUATION**

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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

The surveillance requirements for ECCS alignment provide verification that the assumptions used in the safety analysis are met and that subsystem operability is maintained. 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. With the RHR system crosstie valves closed, the ability of the ECCS to deliver flow to three cold legs following a LOCA coincident with a single failure is impaired. Our evaluation of the impact of this condition is continuing and will be included in a supplement to this report.

IV. CAUSE OF THE EVENT

IMMEDIATE CAUSE

The immediate cause of the event was personnel error resulting in the failure to place the RHR system in the required alignment prior to entry into Mode 3.

LICENSEE EVENT REPORT (LER) **TEXT CONTINUATION**

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V. CORRECTIVE ACTIONS

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.

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.

VII. ADDITIONAL INFORMATION

Additional evaluation is being performed to determine the root causes and contributing factors leading to the event. Corrective actions to prevent recurrence will be based on the results of that evaluation.