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April 11, 1990

William J. Cahill, Jr.
Executive Vice President

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
REACTOR PROTECTION SYSTEM ACTUATION
LICENSEE EVENT REPORT 90-004-00

Gentlemen:

Enclosed is Licensee Event Report 90-004-00 for Comanche Peak Steam Electric Station Unit 1, "Safety Injection Caused by a Failed Blocking Diode." This report also satisfies the reporting requirements of CPSES Technical Specification 3.5.2, (Special Report 1-SR-90-003-00), and 3.4.8.3, (Special Report 1-SR-90-004-00).

Sincerely,

William J. Cahill, Jr.

DEN/daj

Enclosure

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

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1/1

Enclosure to TXX-90140

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)
SAFETY INJECTION CAUSED BY A FAILED BLOCKING DIODE

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

Operating Mode (9): 4
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)	[X] 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)	[X] Other (Specify in Abstract below and in Text, NRC Form 366A) Special Reports 1-SR-90-003-00 1-SR-90-004-00
[ ] 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. NORMAN HOOD SUPERVISOR, COMPLIANCE	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
X	J G	E C B D	W 1 2 0	N					

Supplemental Report Expected (14)

[ ] Yes (If yes, complete Expected Submission Date) [X] No

Expected Submission Date (15): Month | Day | Year

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

On March 12, 1990, with the plant in Hot Shutdown and the Reactor Coolant temperature at 250 degrees Fahrenheit (F) and pressure at 380 Pounds per Square Inch-Gage (PSIG), an Engineered Safety Features Actuation signal occurred which resulted in the initiation of a Train A Safety Injection with flow to the Reactor Coolant System. Control Room personnel responded in accordance with Emergency Operating Procedures. The appropriate offsite agencies were notified of the declaration of the Notification of Unusual Event. During plant recovery, the Technical Specification pressurizer heatup limit was apparently exceeded, and a Reactor Coolant System pressure transient caused a Pressurizer Power Operated Relief Valve to open in response to a Low Temperature Overpressure Protection signal. The plant was restored to a stable condition and offsite agencies were notified of the termination of the Notification of Unusual Event. The Safety Injection actuation was caused by a failed diode on a Safeguards driver board in the Solid State Protection System. Corrective actions include revisions to station procedures to ensure monthly surveillance of the diode and to prevent signal generation during maintenance activities.

I. DESCRIPTION OF EVENT

A. PLANT OPERATING CONDITIONS BEFORE THE EVENT:

On March 12, 1990 at 1400 CST, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 4, Hot Shutdown. The Reactor Coolant System (RCS) (EIIS:(AB)) was at a temperature of 250 degrees Fahrenheit (F) and pressure of 380 Pounds per Square Inch-Gage (PSIG).

B. REPORTABLE EVENT DESCRIPTION (INCLUDING DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES):

Event Classification: An event or condition that resulted in an automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System.

At the time of the event, Radiation Protection technicians (utility, non-licensed) were performing routine maintenance on the Containment Particulate Iodine Gas (PIG) radiation monitor (EIIS:(MON)(IL)). The procedure requires that the Containment PIG monitor be de-energized prior to checking/replacing the filter elements (EIIS:(FLT)(IL)). By design, de-energization of the Containment PIG initiates an Engineered Safety Features Actuation Signal (ESFAS) causing a Containment Ventilation Isolation (CVI) signal. A blocking diode (EIIS:(ECBD)(TG)) is designed to prevent the CVI signal from reaching the Safety Injection relays(EIIS:(RLY)(BQ)). However, the blocking diode had failed in a shorted condition. The signal was not blocked and resulted in the actuation of the Train A Safety Injection relays and the initiation of a Train A Safety Injection.

At 1401 CST numerous alarms were received in the Control Room indicating a Train A Safety Injection actuation; Control Room personnel (utility, licensed) responded in accordance with the Emergency Operating Procedures. All systems and components responded as necessary to perform their safety functions. However, the Motor Driven Auxiliary Feedwater Flow Control Valves (EIIS:(FCV)(BA)) stroked to positions providing uneven flows to Steam Generators (EIIS:(SG)(AB)) one and two.

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

At 1408 CST the Reactor Coolant Pumps (RCP) (EIIS:(P)(AB)) were secured because of fluctuating differential pressures across the number one controlled leakage seals (EIIS:(SEAL)(AB)). At 1414 CST injection flow was terminated, and at 1420 CST a Notification of Unusual Event (NOUE) was declared based on Emergency Core Cooling System (ECCS) (EIIS:(CB)) actuation with flow to the RCS. The NRC was notified via the Emergency Notification System line at 1446 CST in accordance with 10CFR50.72. Additional offsite state and local agencies were notified as well.

During the Safety Injection, approximately 8000 gallons of borated water from the Refueling Water Storage Tank (RWST) (EIIS:(TK)(BE)) was added to the RCS, resulting in an increase in Pressurizer (EIIS:(PZR)(AB)) level. At 1420 CST the Reactor Operator (utility, licensed) re-established letdown flow and reduced charging flow in accordance with the Emergency Operating Procedure. While reducing pressurizer level, an indicated 132 degree F increase in a one hour period exceeded the Technical Specification pressurizer heatup limit.

During plant recovery, Reactor Coolant flow was reinitiated by Control Room personnel (utility, licensed) to establish uniform RCS temperature and to determine the actual RCS temperature. The restart of RCP 4 resulted in a RCS pressure surge. At 1556 CST, the Pressurizer Power Operated Relief Valve (PORV) (EIIS:(PCV)(AB)) actuated in response to a Low Temperature Overpressure Protection (LTOP) signal to mitigate the transient. The maximum pressure reached was 420 PSIG. The system functioned as designed causing a Pressurizer PORV to open for approximately 7 seconds, reducing RCS pressure to 385 PSIG, and preventing the RCS pressure from exceeding the Technical Specification limit of 560 PSIG.

By 1630 CST the plant was restored to a stable condition with RCS temperature at 180 degrees F and pressure at 350 PSIG. Offsite agencies were notified of the termination of the Emergency Class at that time.

C. <u>STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT:</u>

The Containment PIG radiation monitor was removed from service coincident with the Safety Injection actuation.

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APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
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D. **CAUSE OF EACH COMPONENT OR SYSTEM FAILURE, IF KNOWN:**

The failure analysis performed on the diode concluded that this is a random failure, and no root cause can be directly identified for the diode failure.

It could not be determined when the diode failure occurred. On approximately December 14, 1989, a design modification was installed on the Solid State Protection System (SSPS) (EIIS:(JG)) which placed a switch (EIIS:933)(JG)) in series with the subject blocking diode. Prior to entry into Mode 4 the open switch had prevented an ESFAS from passing through the shorted blocking diode. Just prior to entry into Mode 4 this switch was placed in the closed position. When the Containment PIG radiation monitor was de-energized during the first regularly scheduled filter maintenance subsequent to entry into Mode 4, the CVI signal fed into the Safety Injection actuation relays initiating a Train A Safety Injection.

The Safeguards Driver Board containing the failed diode was removed from the SSPS cabinet by I&C Technicians (utility, non-licensed) and tested. The board functioned properly, with the exception of the subject diode, demonstrating that none of the other components on the board contributed to the failure.

A surveillance test was performed on the SSPS cabinets (EIIS:(CAB)(JG)), with a replacement board installed, and all circuits tested correctly. The diode failure was not a result of a malfunction of other circuit boards in the SSPS cabinet.

A review of work history indicates that work was performed in the SSPS cabinet during implementation of a recent design modification. However, no conclusive evidence was identified establishing a relationship between this activity and the diode failure.

Westinghouse visually inspected the card containing the subject diode and found no damage to the card or apparent reason for the diode failure. Westinghouse determined that no other plants have reported a failure of this model diode.

E. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT:

The subject diode did not block the CVI signal from reaching the Safety Injection relays because the diode was failed in a shorted condition, resulting in a Train A Safety Injection.

F. FOR FAILURES OF COMPONENTS WITH MULTIPLE FUNCTIONS, LIST OF SYSTEMS OR SECONDARY FUNCTIONS THAT WERE ALSO AFFECTED:

No failures of components with multiple functions have been identified.

G. FOR FAILURES THAT RENDERED A TRAIN OF A SAFETY SYSTEM INOPERABLE, AN ESTIMATE OF THE ELAPSED TIME FROM THE DISCOVERY OF INOPERABILITY UNTIL THE TRAIN WAS RETURNED TO SERVICE:

The SSPS was not rendered inoperable by the diode failure, but was removed from service to troubleshoot the cause of the event. Following completion of engineering testing activities, maintenance activities to replace the driver board, and performance of logic testing, the SSPS was returned to service on March 14, 1990, at approximately 0900 CST. Train A of the SSPS was unavailable for approximately 42 hours.

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

The Safety Injection actuation resulted in numerous Control Room alarms. Engineering personnel (utility, non-licensed) identified the possible cause of the event at approximately 2300 CST on March 12, 1990; and at 2345 CST, troubleshooting of the SSPS verified the failed diode as the cause of the actuation.

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I. CAUSE DETERMINATION:

SAFETY INJECTION ACTUATION

At the time of the event, Radiation Protection technicians (utility, non-licensed) were performing routine maintenance on the Containment PIG radiation monitor. The procedure requires that the Containment PIG radiation monitor be de-energized prior to checking/replacing the filter elements. By design, de-energization of the Containment PIG initiates an ESFAS causing a CVI signal. A blocking diode is designed to prevent the CVI signal from reaching the Safety Injection relays. However, the blocking diode had failed in a shorted condition. The signal was not blocked and resulted in the actuation of the Train A Safety Injection relays and the initiation of a Train A Safety Injection.

LOW TEMPERATURE OVERPRESSURE PROTECTION ACTUATION

When the decision was made to reinitiate Reactor Coolant flow, Steam Generators 3 and 4 were not being fed with Auxiliary Feedwater(EIIS:(BA)). This allowed the accumulation of a relatively hotter mass of water in those Steam Generators. Train B of Residual Heat Removal (RHR) (EIIS:(BP)) was in service in hot leg recirculation mode with RHR Heat Exchanger (EIIS:(HX)(BP)) at a minimum to reduce the cooldown during the event recovery. RCP 4 was selected for reinitiation of Reactor Coolant flow because of the favorable spray response. When RCP 4 was started, thermal energy was transferred to the relatively cooler water of the RCS, resulting in an increase in RCS inventory volume sufficient to increase system pressure above the LTOP actuation setpoint. Spray response was less effective at controlling pressure than expected because of the presence of non-condensibles in the pressurizer steam space.

MOTOR DRIVEN AUXILIARY FEEDWATER FLOW IMBALANCE

A previous modification designed to prevent pump runout did not consider all system interactions and resulted in uneven flows to the steam generators during this event. Upon receipt of the Safety Injection signal, the flow control valve "Trip-To-Auto" function was initiated. However, the Motor Driven Auxiliary Feedwater Pumps did not sequence on for approximately 30 seconds after receipt of the Safety Injection signal. During the event the flow control valves stroked to positions resulting in uneven flows to Steam Generators 1 and 2.

J. **SAFETY SYSTEM RESPONSES THAT OCCURRED:**

The following safety systems actuated automatically as a result of the event; the appropriate components within these systems operated as designed upon receipt of the Train A Safety Injection actuation signal:

Train A ECCS
Train A Essential Ventilation systems (EIIS:(VF))(EIIS:(VL))
Train A Containment Spray Pumps (EIIS:(P)(BE))
Train A Component Cooling Water (EIIS:(CC))
Train A Motor Driven Auxiliary Feedwater
Control Room HVAC switched to Emergency Recirculation (EIIS:(VI))
Diesel Generator 01 started (EIIS:(EK))
Phase A Containment Isolation (EIIS:(JM))
Feedwater Isolation Valves closed (EIIS:(ISV)(SJ))

K. **FAILED COMPONENT INFORMATION:**

Westinghouse supplied diode CR4 model 1N4148 on safeguards driver board Type 6D69D15G01 WSN0049.

II. **ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS EVENT:**

A. **SAFETY ANALYSIS OF SAFETY INJECTION ACTUATION DURING FULL POWER OPERATIONS:**

An assessment was performed which addresses the safety consequences and implications had this event occurred under the most severe initial conditions of 100% power operation. If diode CR4 was shorted and the Containment PIG radiation monitor was de-energized while operating at full power, a Train "A" Safety Injection Actuation Signal (SIAS) would be generated. As was the case for the actual event, no direct reactor trip would have occurred as a result of this individual SIAS; however, a reactor trip signal would have occurred as a result of the expected transients.

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The event is dominated by the loss of normal feedwater which occurs due to the feedwater isolation. The Loss of Normal Feedwater analysis in the FSAR Chapter 15.2.7 would bound the event for several reasons. (1) The FSAR analysis is performed with assumptions which minimize the decay heat removal capability (i.e., minimum auxiliary feedwater flow following low-low steam generator water level trip and no safety injection). The event has auxiliary feedwater being provided from both motor-driven auxiliary feedwater pumps within a few seconds of the feedwater isolation. This rapid delivery of abundant auxiliary feedwater allows for increased heat removal capability which lessens the severity of the event. (2) In addition, the safety injection provides some benefit by injecting relatively cold borated water into the RCS. The effect of the safety injection would be a decrease in the reactor power prior to the reactor trip, thus resulting in less stored energy in the primary system. The FSAR analyses assume no decrease in reactor power prior to trip. (3) The FSAR analysis is performed at the Engineered Safeguards Power (104.5%). This assumption also conservatively bounds the event at lower powers.

In the long term, the spurious safety injection must eventually be terminated before the pressurizer becomes water solid. However, the reactor operator would have a considerable length of time to perform this function. The FSAR analysis of the spurious safety injection event is performed to demonstrate that the operator has sufficient time to terminate the safety injection prior to filling the pressurizer. The assumptions in that analysis are directed toward maximizing the filling of the pressurizer. The scenario is bounded by the FSAR Spurious Safety Injection analysis for pressurizer overfill.

It is concluded that if diode CR4 was shorted and the Containment PIG radiation monitor was de-energized while operating at full power, the ensuing event would be within the accident analysis of FSAR Sections 15.2.7 and 15.5.1. Therefore, this event would not adversely affect the safe operation of CPSES Unit 1 or the health and safety of the public. The safety significance of the actual event is bounded by the above discussion.

B. **ANALYSIS OF THERMAL TRANSIENT:**

An initial evaluation performed by Westinghouse indicated that the transient had no adverse effect on Pressurizer integrity, RCS loop piping, or the Reactor Vessel (EIIS:(RCT)(AB)). This satisfies the evaluation requirements of CPSES Technical Specification 3.4.8.2. A detailed engineering evaluation, including fatigue and fracture analysis, will be performed by Westinghouse to determine the specific effects of this transient on the design life of the plant.

C. **ANALYSIS OF AUXILIARY FEEDWATER IMBALANCE:**

Reactor Engineering performed an evaluation of the unequal Auxiliary Feedwater flow-split condition discovered during the event. A conservative approach was taken in the evaluation to determine the effect if the event had occurred during normal power operation. It is concluded that the postulated scenario would not result in the plant being in an unanalyzed condition that would significantly compromise plant safety. Furthermore, the postulated scenario is bounded by the conservative assumptions used in the analyses presented in the CPSES FSAR.

III. CORRECTIVE ACTIONS

A. **IMMEDIATE CORRECTIVE ACTIONS:**

A work order was issued to troubleshoot and to replace the defective SSPS card. This was performed in conjunction with an engineering test procedure to confirm the cause of the event and, after replacement of the card containing the failed diode, verify proper operation. Another work order was written to perform logic testing of the SSPS.

B. **ACTIONS TO PREVENT RECURRENCE:**

SAFETY INJECTION ACTUATION
Instrument and Control procedures were changed to include a check that the diode is functioning correctly during the monthly surveillance of SSPS.

The Radiation Protection procedure in use at the time of the event will be revised to prevent signal generation during regularly scheduled maintenance of the Containment PIG radiation monitor.

LTOP ACTUATION
The LTOP setpoints were discussed with Reactor Engineering. The setpoints were recently evaluated by Westinghouse and determined to be satisfactory. Operations Department Administrative Procedures are being revised to incorporate the lessons learned from this event, and Emergency Operating Procedures are being revised to enhance specific criteria for starting RCPs. The lessons learned from this event are being incorporated into the operator training program.

MOTOR DRIVEN AUXILIARY FEEDWATER FLOW IMBALANCE
A design change was implemented to remove the "Trip-to-Auto" initiation resulting from a high-flow and low-pressure condition for the Motor Driven Auxiliary Feedwater Pump flow control valves, and the controllers have been set to take the valves to full open on receipt of the pump actuation start signal. The condition was documented in accordance with Station Administrative Procedures, and is being evaluated as a Plant Incident.

IV. PREVIOUS SIMILAR EVENTS

There have been no previous similar events reported pursuant to 10CFR50.73.

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V. ADDITIONAL INFORMATION:

This report satisfies the reporting requirements of CPSES Technical Specification 3.5.2, which requires that a Special Report be submitted to the Commission within 90 days in the event the ECCS is actuated and injects water into the RCS. Total accumulated actuation cycles to date is one (Special Report 1-SR-90-003-00).

This report satisfies the reporting requirements of CPSES Technical Specification 3.4.8.3, which requires that a Special Report be submited to the Commission within 30 days in the event a Pressurizer PORV is used to mitigate a RCS pressure transient (Special Report 1-SR-90-004-00).