TU ELECTRIC

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

September 13, 1994

C. Lance Terry
Group Vice President

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) - UNIT 2
DOCKET NO. 50-446
ENGINEERED SAFETY FEATURE ACTUATION
LICENSEE EVENT REPORT 446/94-012-00

Gentlemen:

Enclosed is the Licensee Event Report (LER) 94-012-00 for Comanche Peak Steam Electric Station Unit 2 "Manual Reactor Trip and Auxiliary Feedwater Auto-Start due to Steam Generator LO-LO Level Signal."

Sincerely,

NR Blevins

C. L. Terry

OB:bm

ENCLOSURE

cc: Mr. L. J. Callan, Region IV
Mr. D. W. Chamberlain, Region IV
Resident Inspectors, CPSES

P. O. Box 1002 Glen Rose, Texas 76043

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Enclosure to TXX-94237

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)
COMANCHE PEAK-UNIT 2	05000446	1 OF 7

Title (4)

MANUAL REACTOR TRIP AND AUXILIARY FEEDWATER AUTO-START DUE TO STEAM GENERATOR LO-LO LEVEL SIGNAL

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
08	15	94	94	- 012 -	00	09	13	94	N/A	05000
									N/A	05000

Operating Mode (9): 1

Power Level (10): 075

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
W.G. GULDEMOND, MANAGER, SYSTEM ENGINEERING	817-897-8739

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
	SB	ECBD	W	N					

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 August 15, 1994, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1 with reactor power at 75 percent. The unit was in the process of being shut down due to an oil leak on Main Transformer (EIIS:(XFMR)) (2MT2) High Voltage Phase Bushing. At 10:45 a.m. CST the unit was manually tripped to deenergize the main transformer in order to prevent damage to the bushing and/or the transformer. During the unit trip, Reactor Coolant Pump (EIIS:(P)(AB)) (RCP 2-01) tripped due to an electrical transient caused by the Unit trip. Repeated Auxiliary Feedwater pump (EIIS:(BA)) auto-start Engineered Safety Feature (ESF) initiation signals occurred due to Steam Generator (EIIS:(SG)(SB)) LO-LO level signals.

The oil leak was the result of cracking/failure of the bushing housing on 2MT2. The failed bushing assembly has been sent to the vendor for analysis. The ESF actuation occurred as a normal result of the Steam Generator level shrink from the reactor trip following the closure of the turbine stop valves. Subsequent ESF initiation signals occurred due to a failed control card in the Steam Dump control circuit (EIIS:(SB)(ECBD)). The corrective actions were to replace the failed circuit card.

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.

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On August 15, 1994, at approximate 10:45 a.m., Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1, Power Operation, with reactor power at 75 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

On August 14, 1994, at approximately 6:39 a.m. CST, security personnel (contractor, non-licensed) informed the control room staff of oil leaking from the Unit 2 Main Transformer (2MT2). The control room staff (utility, licensed) dispatched electrical maintenance personnel (utility, non-licensed) to investigate the problem. Electrical maintenance personnel reported leakage coming from the 2MT2 High Voltage Phase C bushing and after limited online repair attempts determined that further repairs required an outage.

On August 15, 1994, at approximately 10:40 a.m., a management decision was made to trip the plant. The onshift operations staff were briefed on the trip and recovery. At 10:45 a.m., operations manually tripped the unit from 75 percent power.

Operations implemented their trip recovery procedure. During the electrical bus transfer following the trip, Reactor Coolant Pump 2-01 tripped. Operations responded with the required procedure for the loss of the Reactor Coolant Pump.

The resulting transient from the main turbine trip caused a Steam Generator LO-LO Auxiliary Feedwater actuation. The group 1 Steam Dump valves failed to modulate closed after T_{ave} decreased below the trip open setpoint. The valves closed on the LOW-LOW T_{ave} interlock (P-12). At 10:49 a.m. and 10:53 a.m., the group 1 Steam Dumps cycled open and closed for approximately 15 seconds as the Permissive P-12 cycled. Repeated Steam Generator LO-LO Auxiliary Feedwater (AFW) actuation signals occurred as the Steam Generators experienced shrink and swell caused by the steam dumps cycling. Since the AFW pumps were already in service, they were not effected. At 10:54 a.m., operations placed the steam dump control to the steam pressure control mode, which stopped the level cycles.

No other abnormalities were noted. The plant responded correctly to the plant conditions. Operations completed the trip recovery procedure and placed the plant in Mode 3, Hot Standby.

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

At 10:45 a.m. CST, on August 15, 1994, Unit 2 was tripped from 75 percent power due to oil leak from the main transformer. The ESF actuation was annunciated by numerous alarms in the Control Room.

II. COMPONENT OR SYSTEM FAILURES

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

The outside casting and the inside porcelain were cracked on the failed bushing assembly from 2MT2. There was a blackened area on the porcelain above the flange that may have come from arcing. This appears to be an unusual failure for this component and is considered to be an isolated occurrence.

The steam dump group 1 failed to modulate close and cycled open and close on P-12 signal due to a failed circuit card in the control circuit. The repeated steam dump cycles caused excessive Steam generator shrink and swell which actuated the steam generator LO-LO AFW pump to auto start.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

The cause of the failure of the circuit card is not known.

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**

Manufacturer: Westinghouse
Part: Circuit Card
Model 7300 NSA1
Style 2838A Group 1

III. ANALYSIS OF THE EVENT

A. **SAFETY SYSTEM RESPONSES THAT OCCURRED**

The following safety system actuations occurred as expected as a result of this event.

Manual Reactor trip
Auxiliary Feedwater System (AFW)(EIIS:BA)

B. **DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY**

There were no safety system trains inoperable during this event.

C. **SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT**

The event consisted of a manual reactor trip prompted by the threatened loss of a main transformer. The trip was initiated to preclude any damage to the main transformer. The trip breakers opened as required and all control rods inserted into the core to safely place the plant in Mode 3 Hot Standby with the main transformer de-energized. The loss of Reactor Coolant Pump 2-01 is bounded by the Partial Loss of Forced RCS Flow analysis in FSAR Chapter 15.3.1 that concluded that the DNBR will not decrease below the limit value at any time during the transient.

The failure of the group 1 steam dumps to modulate closed is bounded by the Inadvertent Opening of a Steam Generator Relief or Safety Valve analysis in FSAR chapter 15.1.4 that concluded the DNB design limits are not exceeded.

The August 15, 1994 event did not present a risk to the operation of Comanche Peak Steam Electric Station Unit 2 or the health and safety of the public.

IV. CAUSE OF THE EVENT

A. MANUAL REACTOR TRIP

The root cause was a loss of oil from the main transformer High voltage bushing. Although the oil level indication remained above the Low level limit, TU Electric Transmission Operations Support recommended immediate removal from service. The group based the recommendation on uncertainty of the actual oil level. Experience with sticking indicators on other bushing level gauges created doubt as to the level. The suspect level prompted the manual trip to immediately remove the transformer from service.

Initial inspection by TU Electric Transmission Operation Support personnel of the removed bushing found the outside casting cracked and the internal porcelain cracked.

B. REACTOR COOLANT PUMP 2-01 TRIPPED

Investigation of the Reactor Coolant Pump trip found no problem with the breaker control circuits. The breaker protection relays indicated B & C phase instantaneous over current trips. The most likely cause was the result of the starting current exceeding the instantaneous setpoint when the motor re-energized on the bus transfer. The transfer from the main source (Breaker 2A1-1) to the alternate source (Breaker 2A1-2) is a fast transfer. The alternate breaker starts to close when the normal breaker opens. However, the bus is dead for approximately 9 cycles before the alternate source re-energizes the bus. During this dead bus time, the motors (still connected to the bus and generating back EMF) slow down with voltage (frequency) dropping. When the bus re-energizes, the connected motors draw locked rotor current briefly until they reach full speed again. On the first few cycles, depending on the normal bus

voltage compared to the back EMF voltage and frequency, a significant offset current can develop approaching 11X full-load current. In one to two cycles the current decreases to normal locked rotor current. The instantaneous phase relays are set at 9.12X full-load current. The inrush current may have exceeded the setpoint long enough to actuate the phase B and C phase relays.

C. STEAM DUMP GROUP 1 FAILED TO MODULATE CLOSE AND CYCLED OPEN AND CLOSE ON P-12 SIGNAL

The Steam Dump group 1 failed to modulate causing the steam generator perturbation which generated Auxiliary Feedwater actuation signals on LO-LO Steam generator level. The root cause was the failed circuit card 2-UY/765N in the control circuit. Troubleshooting found relay 2-TY/500E on the card energized when it should have been de-energized . In this condition, permissive P-12 interrupts the air supply to the steam dump group if that steam dump is in T_{ave} mode.

V. CORRECTIVE ACTIONS

A. LOSS OF OIL FROM 2MT2 REQUIRED A MANUAL REACTOR TRIP

TU Electric Transmission replaced the bushing and returned the transformer to service. The failed bushing has been sent to the vendor for failure analysis.

B. REACTOR COOLANT PUMP 2-01 TRIPPED FROM THE BUS TRANSFER

The Reactor Coolant Pump trip during the bus transfer was most likely caused by an anomaly of high inrush current on re-energization of the motor. Engineering review of bus transfer startup test found all transfer times acceptable. Discussion with the Westinghouse Electro-Mechanical Division indicates the motor is designed to handle the stresses generated by re-energization after a short dead time. The trip of Reactor Coolant Pump 2-01 was determined to be an expected response for some portion of the time under these conditions, and requires no further action.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

Facility Name (1)	Docket Number (2)	LER Number (6) Year	Sequential Number	Revision Number	Page (3)
COMANCHE PEAK-UNIT 2	0 5 0 0 0 4 4 6	9 4	0 1 2	0 0	7 OF 7

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

C. STEAM DUMP GROUP 1 FAILED TO MODULATE CLOSE AND CYCLED OPEN AND CLOSE ON P-12 SIGNAL FROM FAILED CIRCUIT CARD 2-UY/765N

The circuit card 2-UY/765N has been replaced.

VI. PREVIOUS SIMILAR EVENTS

There have been no other previous LERs which dealt with a unit trip followed by RCP trip and an Auxiliary Feedwater ESF actuation.