

VISTRA ENERGY



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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Ref 10 CFR 50.73

01/22/2018

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT – UNIT 2
DOCKET NO. 50-446
REACTOR TRIP DUE TO TRIP OF BOTH MFPS

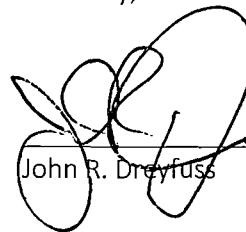
Dear Sir or Madam:

Enclosed is Licensee Event Report (LER) 2-17-003-00, "Unit 2 Reactor Trip Due to Trip of Both MFPS" for Comanche Peak Nuclear Power Plant (CPNPP), Unit 2.

This letter contains no new regulatory commitments.

If you have any questions regarding this submittal, please contact Garry Struble at (254) 897-6628 or garry.struble@luminant.com.

Sincerely,


John R. Dreyfuss

Enclosure COMANCHE PEAK NUCLEAR POWER PLANT – UNIT 2,
REACTOR TRIP DUE TO TRIP OF BOTH MFPS
LICENSEE EVENT REPORT (LER) 2-17-003-00

IEZZ
NRR

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Kriss Kennedy, Region IV
Balwant Singal, NRR
Resident Inspectors, Comanche Peak

**LICENSEE EVENT REPORT (LER)**

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form
<http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/>)

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOF-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME

Comanche Peak Nuclear Power Plant, Unit 2

2. DOCKET NUMBER

05000 446

3. PAGE

1 OF 4

4. TITLE

Manual Reactor Trip due to trip of both Main Feedwater Pumps

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED			
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER		
11	25	2017	2017	003	00	01	22	2018	FACILITY NAME	DOCKET NUMBER		
										05000		
9. OPERATING MODE			11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)									
1			<input type="checkbox"/> 20.2201(b)			<input type="checkbox"/> 20.2203(a)(3)(i)			<input type="checkbox"/> 50.73(a)(2)(ii)(A)		<input type="checkbox"/> 50.73(a)(2)(viii)(A)	
			<input type="checkbox"/> 20.2201(d)			<input type="checkbox"/> 20.2203(a)(3)(ii)			<input type="checkbox"/> 50.73(a)(2)(ii)(B)		<input type="checkbox"/> 50.73(a)(2)(viii)(B)	
			<input type="checkbox"/> 20.2203(a)(1)			<input type="checkbox"/> 20.2203(a)(4)			<input type="checkbox"/> 50.73(a)(2)(iii)		<input type="checkbox"/> 50.73(a)(2)(ix)(A)	
			<input type="checkbox"/> 20.2203(a)(2)(i)			<input type="checkbox"/> 50.36(c)(1)(i)(A)			<input checked="" type="checkbox"/> 50.73(a)(2)(iv)(A)		<input type="checkbox"/> 50.73(a)(2)(x)	
10. POWER LEVEL 100			<input type="checkbox"/> 20.2203(a)(2)(ii)			<input type="checkbox"/> 50.36(c)(1)(ii)(A)			<input type="checkbox"/> 50.73(a)(2)(v)(A)		<input type="checkbox"/> 73.71(a)(4)	
			<input type="checkbox"/> 20.2203(a)(2)(iii)			<input type="checkbox"/> 50.36(c)(2)			<input type="checkbox"/> 50.73(a)(2)(v)(B)		<input type="checkbox"/> 73.71(a)(5)	
			<input type="checkbox"/> 20.2203(a)(2)(iv)			<input type="checkbox"/> 50.46(a)(3)(ii)			<input type="checkbox"/> 50.73(a)(2)(v)(C)		<input type="checkbox"/> 73.77(a)(1)	
			<input type="checkbox"/> 20.2203(a)(2)(v)			<input type="checkbox"/> 50.73(a)(2)(i)(A)			<input type="checkbox"/> 50.73(a)(2)(v)(D)		<input type="checkbox"/> 73.77(a)(2)(i)	
			<input type="checkbox"/> 20.2203(a)(2)(vi)			<input type="checkbox"/> 50.73(a)(2)(i)(B)			<input type="checkbox"/> 50.73(a)(2)(vii)		<input type="checkbox"/> 73.77(a)(2)(ii)	
			<input type="checkbox"/> 50.73(a)(2)(i)(C)			<input type="checkbox"/> OTHER		Specify in Abstract below or in NRC Form 366A				

12. LICENSEE CONTACT FOR THIS LERLICENSEE CONTACT
Garry W StrubleTELEPHONE NUMBER (Include Area Code)
(254) 897-6628**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE) ☒ NO**15. EXPECTED SUBMISSION DATE**

MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On November 25, 2017 Comanche Peak, Unit 2 received alarms indicating a trip of both main feedwater pumps. After confirming a decreasing water level in all four steam generators, the control room initiated a manual reactor trip. All safety systems responded as designed including the automatic start of the auxiliary feedwater system. The cause of the trip of both main feedwater pumps could not be positively identified. Causal analysis indicates that a prior plant modification maintained power to abandoned relays in the Solid State Protection System that may have caused both main feedwater pumps to trip. Subsequent actions were taken to remove the fuses that provided power to the abandoned relays on both Unit 1 and Unit 2 to eliminate recurrence from this possible source. Additional corrective actions have been entered into the Comanche Peak Corrective Action Program.

All times below are in Central Standard Time (CST).

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

(See NUREG-1022, R.3 for instruction and guidance for completing this form
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1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Comanche Peak Nuclear Power Plant, Unit 2	05000-446	2017	003	00

NARRATIVE**I. DESCRIPTION OF REPORTABLE EVENT**

At time 2023 on November 25, 2017 Comanche Peak, Unit 2 received alarms indicating a trip of both main feedwater pumps. After confirming a decreasing water level in all four steam generators, the control room initiated a manual reactor trip. The auxiliary feedwater system automatically started as expected.

A. REPORTABLE EVENT CLASSIFICATION

The event is reportable under 10 CFR 50.73(a)(2)(iv)(A), "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section." The system which was manually actuated was the Reactor Protection System (RPS). The Auxiliary Feedwater System (AFW) automatically started as designed due to low-low steam generator water level following the trip.

B. PLANT CONDITION PRIOR TO EVENT

At 2023 on November 25, 2017 Comanche Peak, Unit 2 was operating in MODE 1 at approximately 100% rated thermal power.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND CONTRIBUTED TO THE EVENT

There were no structures, systems, or components which were inoperable prior to the event which contributed to the event. Prior to the trip of both main feedwater pumps, main feedwater was performing its design function.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

At time 0042 on November 25, 2017 Comanche Peak, Unit 2 entered Technical Specification 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation to perform Train A surveillance testing. While performing the test a push-to-test switch failed. Analysis determined the alignment of Train A Solid State Protection System and the failure of the test switch did not cause or contribute to the trip of both main feedwater pumps. The failed switch was replaced on November 26, 2017 and the Solid State Protection System returned to its normal alignment.

At time 2023 on November 25, 2017 Comanche Peak, Unit 2 received alarms 2-ALB-7B FWPT A TRIP and 2-ALB-8A FWPT B TRIP indicating a trip of both main feedwater pumps [EIS:(SJ)(P)]. After confirming a decreasing water level in all four steam generators, the control room initiated a manual reactor trip 14 seconds after the feed pumps tripped. All safety systems responded as designed.

The cause is not determined because there is no evidence that directly confirms why the main feedwater pumps tripped. The initiating cause was not a permanent condition and the trip signal was gone once investigation into the cause of the trip began. Main feedwater pump, main feedwater pump control system (Mark V), and Solid State Protection System diagnosis and analysis were performed with no direct cause for trip of both main feedwater pumps identified. The reactor was started up on November 29, 2017.

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CONTINUATION SHEET**

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				2017	003	00

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

Initial indication of both main feedwater pumps tripping was provided to the Control room operator by annunciated alarms. Operators (Utility, Licensed) confirmed main feedwater pump trips by observing decreasing level in all four steam generators. The reactor was manually tripped approximately 14 seconds after both main feedwater pumps tripped (times as indicated by the plant computer).

II. COMPONENT OR SYSTEM FAILURES**A. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE**

No definitive component or system failure has been identified. A likely cause of the trip of both main feedwater pumps is a possible spurious actuation of abandoned relays in the Solid State Protection System. A modification performed in 1992 removed the input signals for two water hammer interlocks (steam generator low pressure and low level). The modification did not remove the relays or remove power to relays that previously would trip both main feedwater pumps. The fuses providing power to the abandoned relays were removed on both units. Subsequent corrective actions will remove relays and contacts from the system.

B. FAILURE MODE, MECHANISM, AND EFFECTS OF EACH FAILED COMPONENT

Although both main feedwater pumps tripped, it cannot be positively confirmed what device actuated to cause the trips. Since both main feedwater pumps tripped at the same time, the circuits that could have caused the trip are limited to the Solid State Protection System auxiliary relays or some malfunction of the Mark V Digital feedwater control system that caused the processors to initiate a trip of both main feedwater pumps. The most likely (or least unlikely) is a spurious actuation of an abandoned relay because an actuation of either train relay can trip both main feedwater pumps without a trip of the Main Turbine and this is the only logical component that requires a single circuit failure to cause the event as it occurred on November 25, 2017.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF COMPONENTS WITH MULTIPLE FUNCTIONS

This event did not involve systems or secondary functions which were affected by the possible spurious relay actuation.

D. FAILED COMPONENT INFORMATION

No specific component was determined to be failed.

III. ANALYSIS OF THE EVENT**A. SAFETY SYSTEM RESPONSES THAT OCCURRED**

The Reactor Protection System responded as designed to the manual trip input by the plant operators. All plant safety systems responded as designed. Automatic start of the AFW system was the expected response and the system responded as designed.

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Comanche Peak Nuclear Power Plant, Unit 2	05000- 446	<table border="1"><tr><th data-bbox="1122 370 1203 412">YEAR</th><th data-bbox="1243 370 1382 412">SEQUENTIAL NUMBER</th><th data-bbox="1438 370 1503 412">REV NO.</th></tr><tr><td data-bbox="1122 412 1203 455">2017</td><td data-bbox="1243 412 1382 455">003</td><td data-bbox="1438 412 1503 455">00</td></tr></table>	YEAR	SEQUENTIAL NUMBER	REV NO.	2017	003	00
YEAR	SEQUENTIAL NUMBER	REV NO.						
2017	003	00						

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

The event reported herein did not involve the inoperability of any safety system component or system.

Unit 2, Train A Solid State Protection System was in a Slave Relay Testing alignment for the surveillance test with the failed test switch described above. This alignment did not affect the trip of both main feedwater pump event or the possible spurious relay actuation.

Unit 2, Train A Solid State Protection System was inoperable for 27 hours and 3 minutes.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

A loss of normal feedwater is an ANS Condition II event (Faults of Moderate Frequency). When both main feedwater pumps tripped the reactor was manually tripped and the auxiliary feedwater system automatically started to provide feedwater to the steam generators. The reactor trip on low-low level in any steam generator provides the necessary protection against a loss of normal feedwater.

No automatic safety functions were exercised other than the expected automatic start of the Auxiliary Feedwater System and all plant safety systems responded as designed during the resultant transient. This event had no impact on nuclear safety, reactor safety, radiological safety, environmental safety or the safety of the public. This event has been evaluated to not meet the definition of a safety system functional failure per 10 CFR 50.73(a)(2)(v).

IV. CAUSE OF THE EVENT

A definitive cause for the trip of both main feedwater pumps was not identified. Any failure of the main feedwater pumps or the digital feedwater control system (Mark V) that could trip both main feedwater pumps were eliminated. Several potential main feedwater pump trip signals from the Solid State Protection System were eliminated. However, a possible cause is the spurious actuation of an abandoned relay from a previous plant modification. Inputs to the abandoned relays were removed but the relays were abandoned in place with power available to the relay circuit. The abandoned relays and associated contacts have not been cycled by operation or testing for approximately 25 years.

V. CORRECTIVE ACTIONS

The main feedwater pumps, the digital feedwater control system (Mark V), and the Solid State Protection System were diagnosed and analyzed to determine a cause. The main feedwater pumps and the digital feedwater control system were tested and no failures were identified. The Solid State Protection System analysis identified a possible cause in a spurious actuation of abandoned relays. The power fuses for these relays were removed on both Unit 1 and Unit 2. All proposed activities initiated as a result of this event are being tracked and managed in the Comanche Peak Corrective Action Program.

VI. PREVIOUS SIMILAR EVENTS

There have been no similar reportable events at Comanche Peak in the past three years.