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CP-201800010
TXX-18002

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
DOCKET NO. 50-446
AUXILIARY FEEDWATER SYSTEM ACTUATION DURING UNIT 2 TURBINE TRIP
LICENSEE EVENT REPORT 446/17-001-01

Dear Sir or Madam:

Pursuant to 10CFR50.73, Vistra Operations Company LLC (Vistra OpCo), hereby submits enclosed Licensee Event Report 446/17-001-01, "Auxiliary Feedwater System Actuation During Unit 2 Turbine Trip" for Comanche Peak Nuclear Power Plant (CPNPP) Unit 2.

This communication contains no new licensing basis commitments regarding CPNPP Units 1 and 2.

If you have any questions regarding this submittal, please contact Gary L. Merka at 254-897-6613.

Sincerely,

John R. Dreyfuss

IE22
NRR

Enclosure

c - Kriss Kennedy, Region IV
Margaret W. O'Banion, NRR
Resident Inspectors, Comanche Peak

NRC FORM 366 (04-2017) | U.S. NUCLEAR REGULATORY COMMISSION

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/)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Comanche Peak Nuclear Power Plant	05000 446	1 OF 3

4. TITLE

Auxiliary Feedwater System Actuation During Unit 2 Turbine Trip

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 05000
08	11	2017	2017 -	001 -	01	01	22	2018	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 10

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☑ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
	☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Timothy A. Hope, Manager, Regulatory Affairs	6370

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
X	SJ	FCV	Fisher	Y					

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

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

At 1124 Central Daylight Time on August 11, 2017, Comanche Peak Nuclear Power Plant (CPNPP) Unit 2 experienced an automatic Auxiliary Feedwater System actuation during a Turbine trip. The plant was stabilized at 3 percent reactor power with the Auxiliary Feedwater System feeding all Steam Generators with all levels within their normal bands. The cause of the Turbine trip was high water level in Steam Generator 2-02 related to the mechanical malfunction of a Steam Generator 2-02 flow control bypass valve. The valve malfunctioned due to a loose locknut on the valve hand wheel. Corrective actions included repair of the Steam Generator 2-02 flow control bypass valve. All times in this report are approximate and Central Daylight Time unless noted otherwise.

NRC FORM 366A (04-2017) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

(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/)

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO.
Comanche Peak Nuclear Power Plant	05000- 446	17	001	01

NARRATIVE

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

This event is reportable under 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 that actuated was the Unit 2 Auxiliary Feedwater System.

B. PLANT CONDITION PRIOR TO EVENT

At 1124 on August 11, 2017, Comanche Peak Nuclear Power Plant (CPNPP) Unit 2 was in MODE 1 operating at approximately 10% power while increasing power following a refueling outage.

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 11, 2017, CPNPP Unit 2 was in MODE 1 operating at approximately 10% power while increasing power following a refueling outage. After syncing the Main Generator to the grid at 1120, Operators (utility, licensed) in the Unit 2 Control Room noted increasing water level in Steam Generator (SG) 2-02. The SG 2-02 flow control bypass valve [EIIS: (SJ)(FCV)] was demanded closed, but the valve would not close and remained in mid-position. Operators then attempted to close the valve via the hand switch on the Main Control Board, but SG 2-02 water level continued rising, and at 1124 Unit 2 received a P-14 signal resulting in an automatic Turbine trip, a trip of the 2B Main Feedwater Pump, a Feedwater Isolation signal, and an automatic Auxiliary Feedwater (AFW) pump start as designed. All systems responded normally during and following the Turbine trip.

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

Operators (utility, licensed) in the Unit 2 Control Room noted increasing level in Steam Generator (SG) 2-02 followed by an automatic Turbine trip on a P-14 signal.

II. COMPONENT OR SYSTEM FAILURES

A. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

The cause of the SG 2-02 flow control bypass valve malfunction was a loose locknut on the valve hand wheel.

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

The cause of the SG 2-02 flow control bypass valve malfunction was a loose locknut on the valve hand wheel. The loose locknut backed off and prevented the valve from going fully closed.

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

Below 25 percent load, the SG 2-02 flow control bypass valves automatically maintain the steam generator water level by using control signals from the Steam Generator water levels. The valves are air operated and are designed to fail closed upon loss of air.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

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1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO.
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NARRATIVE

D. FAILED COMPONENT INFORMATION
The SG 2-02 flow control bypass valves are 4 inch, carbon steel, globe valves. The valves are a Model ED manufactured by Fisher Controls.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED
The Turbine Trip and Auxiliary Feedwater Systems actuated as required.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY
Not applicable - No safety system train inoperability resulted from this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT
This event is bounded by the CPNPP Final Safety Analysis Report (FSAR) accident analysis which assumes conservative initial conditions which bound the plant operating range and other assumptions which could reduce the capability of safety systems to mitigate the consequences of the transient. Feedwater system malfunctions that result in an increase in feedwater flow are analyzed in section 15.1.2 of the FSAR. The system is analyzed to demonstrate plant behavior in the event that an excessive feedwater addition occurs due to a control system malfunction or operator error. The FSAR analysis shows that the departure from nucleate boiling ratio encountered for an excessive feedwater addition at power is above the limit value and the feedwater malfunction event at no-load is bounded by the feedwater malfunction event at full power. The event of August 11, 2017, occurred at 10% reactor power, and all safety systems and components functioned as designed. Based on the above, it is concluded that the health and safety of the public were unaffected by this condition and this event has been evaluated to not meet the definition of a safety system functional failure per 10CFR50.73(a)(2)(v).

IV. CAUSE OF THE EVENT
The AFW actuation was caused by a P-14 signal that was received due to high level in SG 2-02 related to the mechanical malfunction of a Steam Generator 2-02 flow control bypass valve. The cause of the SG 2-02 flow control bypass valve malfunction was a loose locknut on the valve hand wheel. During the Unit 2 15th refueling outage, maintenance was performed on the Steam Generator 2-02 flow control bypass valve. During reassembly of the valve, the maintenance procedure step for tightening the hand wheel locknut at that time did not specify a torque requirement or verification. It is believed that the locknut was not tightened sufficiently in 2RF15, and the loose locknut subsequently backed off and prevented the valve from going fully closed during the August 11, 2017 Unit 2 Turbine trip.

V. CORRECTIVE ACTIONS
The SG 2-02 feedwater flow control bypass valve was repaired. An extent of condition review was performed and verified that all of the other flow control bypass valves were closed. The applicable maintenance procedure has been revised to add a “snug tight” torque value to the lock nut. Per the CPNPP Corrective Action Program, actions will be taken to either add a chain and lock to the hand wheel or a preventive maintenance activity will be created to verify lock nut tightness prior to every start-up.

VI. PREVIOUS SIMILAR EVENTS
A similar reportable event occurred at CPNPP on October 3, 2015 related to a feedwater flow control valve malfunction (Unit 2 LER 446/15-002). The cause of that event was a degraded positioner O-ring, and the details/causes of the August 11, 2017 event are believed to be sufficiently different from the October 3, 2015 event such that the previous corrective actions could not have prevented this event.