NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fort Calhoun Station Unit No. 1	0 5 0 0 0 2 8 5	1 OF 0 3

TITLE (4)
Failure of Main Steam Safety Valve to Lift Within Setpoint Tolerance

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 3	0 3	8 4	8 4	– 0 0 2 –	0 0	0 4	0 2	8 4	N	0 5 0 0 0
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OPERATING MODE (9): 3

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)	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)	X 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)	X 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
Merl R. Core Supervisor-Maintenance Fort Calhoun Station	AREA CODE 4 0 2 4 2 6 - 4 0 1 1

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	S B	R V	D 2 4 3	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

While performing surveillance test ST-MSSV-1, Main Steam Safety Valve Test, during a scheduled shutdown of the unit for refueling, it was discovered that five of the ten Main Steam Safety Valves failed to lift within plus or minus one percent of their nameplate setpoint values. This exceeds the minimum operability requirements of Technical Specification 2.1.6(3) which requires eight of the ten steam safety valves to be operable with their lift settings between 1000 psia and 1050 psia with a tolerance of plus or minus one percent of the nominal nameplate setpoint values whenever the reactor is in power operation.

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

During routine surveillance testing, it was discovered that the lift settings of five of the ten Main Steam Safety Valves were not within plus or minus one percent of their nameplate ratings. This exceeds the minimum operability requirements of Technical Specification 2.1.6(3). The five valves which failed the test were MS-275 and MS-278 associated with Steam Generator A and MS-280, MS-281, and MS-282 associated with Steam Generator B. All five valves, which are Dresser Industries maxiflow model, 6 inch, 1500 psig class relief valves, were recalibrated to within plus or minus one percent of nameplate setpoint values and then retested twice to prove operability and repeatability within three hours of the time of discovery of inoperability. Below is a summary of the as found and as left condition of each valve. MS-275 set pressure is 1035 psig; as found pressure was 1052 psig and as left pressure was 1026 psig. MS-278 set pressure is 1000 psig; as found pressure was 976 psig and as left pressure was 997 psig. MS-280 set pressure is 1025 psig; as found pressure was 1064 psig and as left pressure was 1031 psig. MS-281 set pressure is 1010 psig; as found pressure was 961 psig and as left pressure was 1013 psig. MS-282 set pressure is 1000 psig; as found pressure was 940 psig and as left pressure was 1005 psig. In conversations with Dresser Industries personnel, it has been determined that the method used to test safety valves at Fort Calhoun Station is correct. However, Dresser commented that the valves should be allowed to cool 10 to 15 minutes after the second lift. This practice is observed at Fort Calhoun; however, is not written in the test. ST-MSSV-1 will be revised to include this comment and the valves will continue to be tested using ST-MSSV-1 each refueling outage.

The test was performed during a scheduled shutdown of the unit for refueling. The plant was in a hot shutdown condition, Mode 3, at the time of discovery. The testing of the safety valves was performed between 0500 and 1100 hours on March 3, 1984. The cause is attributed to normal drift of the safety valve lift setting over an operating cycle. The IEEE Std. 803-1983 component function identifier is RV, Relief Valve; the system name is SB, Main/Reheat Steam System.

Overpressure protection at the Fort Calhoun Station is ensured by means of primary safety valves, secondary safety valves, and the reactor protective system. The worst case pressure transient, loss-of-load, in conjunction with a delayed reactor trip, is the design basis for determining the adequacy of the Fort Calhoun Station safety valves. The primary safety valves, secondary safety valves, and reactor protective system maintain reactor coolant system and steam generator pressures below 110% of their respective design pressures during worst case transients.

An analysis of the loss-of-load (LOL) event demonstrates that there is additional secondary system safety valve and primary system safety valve capacity above that which is required to provide overpressure protection. The analysis shows that the two sets of secondary system safety valves with the highest setpoint pressure never achieve a full open position and that the higher setpoint primary system safety valves never open during the design basis transient.

The expected effect of this safety valve setpoint out-of-tolerance condition would be as follows. The safety valves which had opening pressures lower than required would have achieved a full open position sooner and would have released more steam than was calculated in the original LOL analysis. This effect would be to lower the secondary pressures. The valves which had an opening pressure greater than required would have achieved a full open position later than was assumed in the LOL analysis. However, this slower opening would have been more than compensated for by the earlier opening of the

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

valves with lower than required lift settings. It is judged that an overpressurization of the secondary system would not have occurred because three of the subject safety valves would have opened earlier due to their lower opening pressures and four of the other safety valves which did not achieve a full open position in the original LOL analysis would have achieved a more fully open position and released more steam. The original LOL analysis showed a peak pressure of at least 40 psia less than the secondary system design pressure, and the District's analysis for this postulated event is still bounded by this conservative value. It is judged that an overpressurization of the primary system would not have occurred because only one of the two primary system safety valves is shown to open in the LOL analysis.

This is the fourth reportable event of this type in which the Main Steam Safety Valves failed to meet the minimum operability requirement of Technical Specification 2.1.6(3). The other three were LER's 82-020, 76-19, and 77-24.

Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

April 2, 1984
FC-121-84
LIC-84-088

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

Reference: Docket No. 50-285

Gentlemen:

Licensee Event Report
for the Fort Calhoun Station

Please find attached Licensee Event Report 84-002 dated April 2, 1984. This report is being submitted per requirements of 10 CFR 50.73.

Sincerely,

W C Jones

W. C. Jones
Division Manager
Production Operations

WCJ/JCB:jmm

Attachment

cc: Mr. Richard P. Denise, Director
Division of Resident, Reactor Project
& Engineering Programs
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

INPO Records Center
Mr. E. G. Tourigny, Project Manager

SARC Chairman
PRC Chairman
Mr. L. A. Yandell, Senior Resident
Inspector
Fort Calhoun File (2)

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