NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
St. Lucie Unit 2	0 5 0 0 0 3 8 9	1 OF 0 3

TITLE (4)

Pressurizer Code Safeties Inoperable Due to Lift Setpoint Drift.

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 4	1 7	8 6	8 6	0 0 7	0 0	0 5	1 9	8 6	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
6	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10) 0 0	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)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
	20.405(a)(1)(iv)	X 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
Duane Mumper- Shift Technical Advisor	AREA CODE 3 0 5 — 4 6 5 - 3 5 5 0

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	A B	R V	C 7 1 0	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 shutdown for normal refueling on April 17, 1986 two of the three pressurizer code safeties (AB) failed to lift within the required pressure range when bench tested. The lift point of the third valve could not be determined due to excessive seat leakage.

Valve V-1201 was found to have a ruptured bellows that allowed boric acid in the reactor coolant system (RCS) to cause corrosion that resulted in binding that increased the lift setpoint. Valve V-1200 was only 14 psi out of tolerance. No cause could be determined other than normal useage. The seats on valve V-1202 were steam cut so that the test bench did not have enough capacity to test the valve.

The power operated relief valves normally provide the RCS with over pressure protection and prevent lifting of the code safeties.

The reactor vessel head was removed so there were no immediate corrective actions. All pressurizer code safety valves were rebuilt and satisfactorily bench tested before the reactor head was reinstalled.

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

DESCRIPTION OF EVENT

On April 17, 1986 during a normally scheduled outage, with the reactor vessel head removed, two out of three pressurizer code safety relief valves (AB) failed to lift within the required pressure of Technical Specification 3.4.2.2. The lift setting of the third valve could not be determined due to excessive seat leakage. The required lift setting for the pressurizer code safeties is 2500 psia +/-1% at normal operating temperature. At test temperature the required lift setpoint is 2515 psi +/-1%. Valve V-1200 lifted at 2554 psi, V-1201 lifted at 2893 psi, and V-1202 leaked too much to obtain a lift setpoint.

The plant normally rebuilds all three of these valves every refueling outage. This is more than is required by the ASME BOILER and PRESSURE VESSEL CODE, Section XI. The ASME code requires only testing one valve as long as that valve passes.

This failure is a violation of Technical Specification 3.4.2.2 and thus is reportable under 10 CFR 50.73 (a) (2) (i) and 10 CFR 50.73 (a) (2) (ii).

CAUSE OF EVENT

When V-1201 was disassembled it was found that the bellows was ruptured. This allowed boric acid in the RCS to get at the valve internals. The boric acid caused extensive corrosion of some of the carbon steel components. This corrosion caused excessive binding that resulted in the lift setpoint being too high.

The setpoint of V-1200 was only 14 psi out of tolerance. No cause was found other than normal drift over the course of an 18 month cycle.

Due to steam cutting of the seats of V-1202 the bench test rig did not have enough capacity to overcome the seat leakage. This prevented obtaining an as found reading before the valve was torn down for rebuilding. The failure modes of each valve appear to be unrelated to each other.

ANALYSIS OF EVENT

According to the St. Lucie Final Safety Analysis Report the Power Operated Relief Valves (PORV) are sized to mitigate all design over pressurization transients and thus avoid challenges to the pressurizer code safety valves. Therefore, the PORV's provided an independent means of over pressure protection while the code safety valves were in a degraded condition. Furthermore it should be noted that for V-1200 the setpoint error was minor (+14 psia). For V-1202 the observed degradation was in the conservative direction, i.e. excessive leakage at test pressure.

CORRECTIVE ACTIONS

No immediate corrective actions were required since the reactor vessel head was already removed when the valves were tested and thus there was no requirement for the safeties. When the first valve failed to lift, additional testing of the safety valves was conducted as required by the ASME code. All of the pressurizer code safety valves were rebuilt and bench tested satisfactorily before the reactor vessel head was replaced.

ADDITIONAL INFORMATION

The pressurizer code safety valves are manufactured by Crosby valve, type NB-BP-86. This is the first event of this type at the St. Lucie Plant, no previous LERs.

FPL

FLORIDA POWER & LIGHT COMPANY

MAY 1 9 1986

1-36-212

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

Gentlemen:

RE: **Reportable Event 86-7**
St. Lucie Unit 2
Date of Event: April 17, 1986
Pressurizer Code Safeties Inoperable Due to Lift Setpoint Drift

The attached Licensee Event Report is being submitted pursuant to the requirements of 10 CFR to provide notification of the subject event.

Very truly yours,

C. O. Woody
Group Vice President
Nuclear Energy

COW/SAV/eh

Attachment

cc: Dr. J. Nelson Grace, Region II, USNRC
Harold F. Reis, Esquire
File 933.1
PNS-LI-86-180

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