Florida Power
CORPORATION

Crystal River Unit 3
Docket No. 50-302

September 21, 1992
3F0992-16

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

Subject: Licensee Event Report (LER) 92-018

Dear Sir:

Enclosed is Licensee Event Report (LER) 92-018 which is submitted in accordance with 10 CFR 50.73.

Sincerely,

Gary Boldt

G. L. Boldt
Vice President
Nuclear Production

EEF:mag

Enclosure

xc: Regional Administrator, Region II
Project Manager, NRR
Senior Resident Inspector

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IE22

NRC FORM 366 (6-89) 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 HOURS. 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)
CRYSTAL RIVER UNIT 3 (CR-3)	0 5 0 0 0 3 0 2	1 OF 0 4

TITLE (4) Safety Valve Test Results Lead To A Condition Prohibited By Technical Specifications

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)
									N/A	0 5 0 0 0
0 8	2 0	9 2	9 2	— 0 1 8	— 0 0	0 9	2 1	9 2	N/A	0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)		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
W. A. Stephenson, Nuclear Safety Supervisor	AREA CODE 9 0 4 7 9 5 - 6 4 8 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	

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)

On July 31, 1992, Crystal River Unit 3 was operating in MODE 1, power operations, at 100% rated thermal power (RTP). Florida Power Corporation discovered that the required testing of pressurizer code safety valves had not been completed On August 18, 1992, the required test results were reported and it was determined that the testing deficiency had led to a condition prohibited by plant Technical Specifications.

The valve tested failed to meet the pressure lift setpoint tolerance during the "as-found" testing. This "as-found" test failure would have required testing both of the code safeties. Because the testing was not performed until after the resumption of power operations, the required testing of the companion safety valve was not performed. The failure to test the companion safety valve constitutes a condition prohibited by Technical Specifications. The cause of this event is personnel error. The two pressurizer safety valves will be removed and tested during the next mid-cycle maintenance outage. An administrative requirement will be developed to assure that both pressurizer safety valves are removed and tested together in all subsequent test periods. A review of the CR-3 valve program will be performed to assure that the requirements of ANSI/ASME OM-1 (1981) standards are being met.

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CRYSTAL RIVER UNIT 3 (CR-3)	0 5 0 0 0 3 0 2	9 2	0 1 8	0 0	0 2 OF 0 4

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

EVENT DESCRIPTION

Florida Power Corporation (FPC) is required by the Crystal River Unit 3 (CR-3) Technical Specification (T.S.) Section 4.4.2 to surveil the pressurizer code safety valves in accordance with the testing requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code and applicable Addenda as required by 10 CFR 50, except where specific written relief has been granted by the NRC. FPC requested relief from the NRC to substitute the American National Standards Institute (ANSI) ANSI/ASME Operations and Maintenance Standard OM-1, 1981 for the 1983 edition of the ASME BPV Code, subsection IWV-3510, which is applicable to testing of the pressurizer code safety valves. This substitution of code requirements changed the testing requirement when it was approved for use by FPC in 1988.

During the mid-cycle 8 maintenance outage which began in October of 1991, one of the two pressurizer code safety valves [AB,RV] was removed for testing. This valve was replaced at the time of removal with a refurbished, qualified valve from the warehouse. Since this test cannot be conducted on site, FPC sends the valves off-site to a vendor for performance of the test. The test result for the code safety valve removed during the mid-cycle outage was received in May 1992, and stated that "as found" set pressure was 2533 pounds per square inch gage (psig) (+0.32% or 8 psig outside the allowable range of 2500 psig plus or minus 1%). During the refuel 8 outage which began in April 1992, the other pressurizer code safety valve was removed. It was also replaced with a warehouse spare. On August 18, 1992, the test result for the second code safety valve was reported to FPC. This result stated that the initial "as found" setpoint pressure was 2534 psig (+0.36% or 9 psig outside the allowable range of 2500 psig plus or minus 1%).

On July 31, 1992, CR-3 was operating in MODE 1, power operations, at 100% rated thermal power (RTP). At 1520 hours it was discovered that the testing of the pressurizer code safety valves [AB,RV] required by the ASME BPV Code, Section XI, and the associated relief request, OM-1, had not been completed. The OM-1 code requires the pressurizer code safety valves to be tested prior to the resumption of power operations. This was not accomplished. Further, the valve removed in each outage failed to meet the pressure lift setpoint tolerance during the "as-found" testing. This "as-found" test failure would have required testing both of the code safeties in each of the past outages. Because the testing was not performed until after the resumption of power operations, the required tests of the companion safety valve [AB,RV] were not performed. This failure to test the companion safety valve [AB,RV] constitutes a condition prohibited by T.S., which is reportable under 10CFR50.73(a)(2)(i)(B).

EVENT EVALUATION

This event did not significantly impact nuclear safety. CR-3 T.S. require that a minimum of one pressurizer code safety valve [AB,RV] shall be OPERABLE with a lift setting of 2500 psig ± 1%. This upper pressure is 2500 plus 25 psig, or 2525 psig. The analyses and T.S. also allow one of the two valves to have a higher lift setpoint (up to 5% of the design value) if the first valve's lift setpoint is within the 1% tolerance.

The pressurizer was not adversely impacted because the code safeties are set at the test facility to the proper pressure and tolerance. The length of time the valve has been installed appears to determine whether there will be a drift in tolerance. In both cases, there was always one newly refurbished valve installed; thus, the likelihood of two valves out of tolerance is very low.

The valves are "as-found" tested by an off-site vendor, set correctly, and returned to the site as a warehouse spare. The "as-found" test cycles the valve several times. After each outage, there was one newly installed valve and one previously installed valve. At each test interval, the valve that had been installed for two operating periods (refuel to mid-cycle period and mid-cycle to refuel period) was removed and tested. Each valve exceeded the lift tolerance on the first lift cycle. The valve removed during the mid-cycle 8 outage was below the lift tolerance on the second lift cycle. The valve removed during the refuel 8 outage was within tolerance on the second lift cycle.

On two occasions, the mid-cycle 8 maintenance outage test and the refueling 8 outage test, the pressurizer code safeties exhibited higher than expected lift settings. This means that the pressure in the Reactor Coolant System (RCS) would have risen slightly over the intended setpoint prior to lifting. However, the accident analyses of FSAR Chapter 14 use the ASME required 110% of design pressure (2500 psig) as the limiting condition for maintaining the integrity of the RCS.

The Analysis Basis Document parameter matrix fixes the upper limit for pressurizer code safety lift at 2540 pounds per square inch absolute (psia), that is 2500 psig nominal value plus the 25 psig error band plus the 15 psi conversion from psig to psia. If this upper bound is met then the assumptions of the analyses are valid. In fact, the initial lift setting for the last valve tested exceeded this value by 8 psi.

The Start Up Accident, the Rod Withdrawal at Rated Power Accident, the Steam Line Break Accident, and the Rod Ejection Accident all have peak pressures well below the 110% of design value (equal to 2765 psia). Only in the Loss of Feedwater Accident is the design value approached.

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

The Loss of Feedwater Accident considers two initiating circumstances; one where feedwater is lost due to feedwater pump shutdown, the other where the feedwater is lost through a main feedwater line break. In the case of loss of feedwater through feedwater pump shutdown, the peak pressure is 2574 psia. This peak occurs after the pressurizer code safeties lift at 2515 psia. In the main feedwater line break case, the pressure peak is higher and also occurs after the safeties lift. No study has been conducted to determine the maximum code safety lift pressure that will result in the maximum allowed RCS pressure of 2765 psia.

CAUSE

The cause of this event is personnel error. The ASME Section XI valve program correctly required the valves to be tested prior to the resumption of power operations. However, station implementing instructions did not mention requirements for the pressurizer code safety valves to be tested prior to resumption of power or to remove the companion valve for testing based on the results of testing the first valve.

CORRECTIVE ACTION

The two pressurizer code safeties [AB,RV] will be removed and tested during the upcoming mid-cycle maintenance outage. Administrative guidance will be developed by February 15, 1993 to assure that the requirements of the ASME BPV Code and the OM-1 (1981) requirements for the pressurizer code safety valves are being met.

PREVIOUS SIMILAR OCCURANCES

There are no previous OM-1 code deficiencies and no instances of pressurizer code safeties not meeting the code test criteria in records reviewed back to 1985.