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)
CRYSTAL RIVER UNIT 3	0 5 0 0 0 3 0 2	1 OF 0 2

TITLE (4)
Reactor Building Containment Penetration Not Designed in Accordance with FSAR

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 1	1 3	8 4	8 4	- 0 0 1	- 0 0	0 2	1 0	8 4	N/A	0 5 0 0 0
									N/A	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)		20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.71(c)
0 0 0		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)	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
R.H. Thompson, Engineer-1	AREA CODE 9 0 4 7 9 5 - 3 8 0 2

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	
A	N I H	P I E N I	C 3 1 0	Y							

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 6	1 3	8 4

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

During a refueling outage (May, 1983) the end cap of a penetration in the reactor containment building was incorrectly cut off. Subsequently a plant modification package was issued to replace the end cap. A routine review of the modification package on January 13, 1984, discovered several design specifications that were inconsistent with FSAR commitments. Cognitive personnel error is the cause of this event in that both the design engineer (on contract to Florida Power Corporation) and the verification engineer (a Florida Power Corporation employee) failed to follow applicable engineering procedures. The verification engineer has been retrained on following applicable engineering procedures. The design engineer no longer works for Florida Power Corporation. The results of the local leak rate test that was performed on the penetration (July 2, 1983) and subsequent engineering evaluation (January, 1984) indicate that the end cap will perform its intended safety function under the worst case LOCA conditions and thus justifies continued operation with the as-built penetration until the next refueling outage (March, 1985). An engineering evaluation will be performed to determine if the as-built reactor containment building penetration is adequate for the remainder of plant life or if another modification is required to make the penetration consistent with FSAR commitments.

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

BACKGROUND

In May, 1983, during a refueling outage, the end plate on the reactor building side of spare penetration #353[1] was mistakenly cut off. A plant modification package was issued on May 19, 1983, to replace the end cap on this penetration. The modification was installed on July 5, 1983.

IDENTIFICATION OF EVENT

A routine review of the modification package on January 13, 1984, discovered that contrary to FSAR commitments:

1. ASTM A-36 was specified as the new end plate material (FSAR sections 5.2.2.4 and 5.2.2.4.2 require SA-516 Grade 60, impact tested to SA-30, and having certified mill test reports).

2. All welding was required to be inspected per B31.7 - 1969. (FSAR sections 5.2.2.4.1, 5.2.2.4.3, and 5.2.2.4.4 require the NDE to be per ASME Section III Class B.)

CAUSE OF EVENT

Cognitive personnel error is the cause of this event in that both the design engineer (on contract to Florida Power Corporation) and the verification engineer (a Florida Power employee) failed to follow applicable engineering procedures.

ANALYSIS OF EVENT

An engineering evaluation was performed subsequent to the discovery of the erroneous design specifications. The results of the engineering evaluation and the local leak rate test that was performed on the penetration after the end cap was installed (July 2, 1983) indicate that the end cap will perform its intended safety function under the worst case LOCA conditions, thus justifying continued operation with the as-built penetration until Refuel V (March, 1985). An engineering evaluation will be performed to determine if the as-built reactor containment building penetration is adequate for the remainder of plant life (beyond Refuel V) or if another modification is required to make the penetration consistent with FSAR commitments.

The applicable Safety Related Engineering Procedures governing plant modifications were also reviewed and are considered adequate if properly used.

Several other reactor containment building penetration plant modification packages were reviewed and found to comply with the FSAR. Hence, this event appears to be an isolated occurrence.

CORRECTIVE ACTION

Both the design engineer and the verification engineer had been trained in the use of applicable Safety Related Engineering Procedures governing plant modifications. The verification engineer has been retrained on following applicable engineering procedures. The design engineer no longer works for Florida Power Corporation.

[1]NH, PEN, Chicago Bridge & Iron (C310)

84 FEB 16 A8:58

Florida Power
CORPORATION

February 10, 1984
3F0284-06

Mr. James P. O'Reilly
Regional Administrator, Region II
Office of Inspection & Enforcement
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, GA 30303

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Licensee Event Report No. 84-001

Dear Mr. O'Reilly:

Enclosed is Licensee Event Report No. 84-001 and the attached supplementary information sheet, which are submitted in accordance with 10 CFR 50.73.

Should there be any questions, please contact this office.

Sincerely,

G. R. Westafer

G.R. Westafer
Manager
Nuclear Operations Licensing and Fuel Management

AEF:jcf

Enclosure

cc: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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