FPL

Florida Power & Light Company, 6501 South Ocean Drive, Jensen Beach, FL 34957

June 12, 2001

L-2001-138
10 CFR 50.73

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

Re: St. Lucie Unit 1
Docket No. 50-335
Reportable Event: 2001-003-00
Date of Event: April 14, 2001
Reactor Coolant System Instrument Nozzle Leakage
Caused by Primary Water Stress Corrosion Cracking

The attached Licensee Event Report 2001-003 is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(ii) to provide notification of the subject event.

Florida Power and Light Company (FPL) submitted additional information on the half nozzle replacement of reactor coolant system hot leg nozzle RC-126 to the NRC, including a flaw evaluation, by FPL letter L-2001-131 dated May 24, 2001.

Very truly yours,

D. E. Jernigan
Vice President
St. Lucie Nuclear Plant

DEJ/GRM

Attachment

cc: Regional Administrator, USNRC, Region II
Senior Resident Inspector, USNRC, St. Lucie Nuclear Plant

IE22

NRC FORM 366 (1-2001)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)

Reactor Coolant System Instrument Nozzle Leakage Caused by Primary Water Stress Corrosion Cracking

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 NAME	DOCKET NUMBER
04	14	2001	2001	- 003 -	00	06	12	2001	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	6
POWER LEVEL (10)	000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

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

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
George R. Madden	(561) 467 - 7155

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	AB	NZL	C490	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 15 single-spaced typewritten lines)* **(16)**

On April 14, 2001, Unit 1 was in refueling MODE 6 following shutdown for a scheduled refueling outage. During maintenance to replace fixed insulation with removable insulation at the 3/4-inch 1B hot leg instrument nozzle RC-126, a minor through wall reactor coolant system (RCS) leak was identified at the nozzle. A review of the Alloy 600 nozzle material used at RC-126 indicates that heat NX-0003 was used. There is no industry failure history for this heat; however, Alloy 600 is susceptible to primary water stress corrosion cracking (PWSCC) and has occurred in similar conditions in the RCS hot leg at St. Lucie Unit 1 and six other Combustion Engineering (CE) Nuclear Steam Supply System (NSSS) units. There are nine uses of this heat in Unit 1, including the one that failed at RC-126. Visual inspection of the other eight nozzles of the susceptible heat NX-0003 was completed and no leakage was identified.

Corrective actions include replacement of RCS nozzle RC-126 under ASME Section XI IWA-7000 with a half nozzle design prior to startup. The RCS inspection procedures were reviewed and are effective at identifying the leakage associated with the small-bore PWSCC failures and no changes were required. FPL is developing a replacement schedule for the remaining eight nozzles.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Description of the Event

On April 14, 2001, Unit 1 was in refueling MODE 6 following shutdown for a scheduled refueling outage. During maintenance to replace fixed insulation with removable insulation at the 3/4-inch 1B hot leg instrument nozzle RC-126, a through wall reactor coolant system (RCS)(EIS:AB) leak was identified at the nozzle. No leakage was noted at instrument nozzle RC-126 during the previous initial plant cooldown and cold shutdown inspections performed in accordance with appendices C and B of plant procedure OP 1-0120022, "Reactor Coolant System Leak Test."

The hot leg instrument nozzle that penetrates the 1B hot leg at RC-126 is an Alloy 600 nozzle. St. Lucie has previously experienced primary water stress corrosion cracking (PWSCC) in Alloy 600 nozzles. In 1995, a similar crack and leak of an RCS hot leg nozzle occurred in St. Lucie Unit 2 (Ref. LER 389/95-004). To date, leakage of Alloy 600 material caused by PWSCC has occurred in over 87 Alloy 600 nozzle or heater sleeves built by Combustion Engineering (CE). The problem has been documented in several Combustion Engineering Owners Group (CEOG) reports.

Cause of the Event

A review of the Alloy 600 nozzle material used at RC-126 indicates that heat NX-0003 was used. There is no industry failure history for this heat; however, Alloy 600 is susceptible to PWSCC and cracking has occurred in similar conditions in the RCS hot leg. There are nine uses of this heat in Unit 1, including the one that failed at RC-126. Inspection of the other eight nozzles of the susceptible heat NX-0003 was completed and no leakage was identified.

Prior to the removal of the 1B leaking hot leg nozzle connected to RC-126, an eddy current testing (ECT) plan to identify the location and orientation of the indications was developed. A 0.50-inch ID ECT probe for the 3/4-inch hot leg nozzle was used to determine relative position and orientation. Based on the ECT results, there were two major flaws that were generally axial with some smaller indications associated with the major flaws located at the inboard welded end. The ECT results revealed that the major indications were approximately 0.83 inches and 0.41 inches long extending from the inboard end of the pipe adjacent to the J-weld attaching the nozzle to the inside diameter of the hot leg pipe. The ECT scan was also performed for circumferential indications but none were identified. The problem has been well documented in industry reports.

There is no safety concern with ejection since the Alloy 600 material is tough and the cracking aligns axial to the nozzle, adjacent to the vessel or (hot leg) pipe inside diameter attachment J-weld. PWSCC cracking characteristics in these types of nozzles are axial cracks in the Alloy 600 nozzle material in the area adjacent to the J-weld. There have been no identified circumferential cracks in the small-bore penetrations at CE NSSS plants. The axial cracking and absence of circumferential cracking has been confirmed by ECT or other NDE surface methods during the St. Lucie Unit 2 pressurizer instrument failures in 1993, 1994, and the hot leg nozzle failure in 1995. The ECT result from the Unit 1, RC-126 failure provides further evidence that this PWSCC mechanism results in axial cracks in these small-bore Alloy 600 penetrations.

No laboratory analysis was conducted since PWSCC has been well documented and sample removal could jeopardize the repair.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Analysis of the Event

This event is reportable under 10 CFR 50.73(a)(2)(ii)(A) as any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

Analysis of Safety Significance

The leakage at the 1B hot leg nozzle RC-126 is the result of PWSCC of the Alloy 600 instrument nozzle.

Technical Specification 3.4.10, Structural Integrity, is applicable in all MODES. The Technical Specification requires that with the structural integrity of any ASME Code Class 1 component(s) not conforming to requirements, the component must be repaired prior to increasing RCS temperature more than 50°F above the minimum temperature required by NDT considerations.

There is no immediate safety concern with ejection of a nozzle since the Alloy 600 material is tough, the cracking is generally axial (aligned with the nozzle), is slow to propagate, and identifiable leakage would occur long before the crack could grow to a point of instability. The CEOG reports also concluded that the only potential safety issue is if the leakage were to go undetected for a significant period of time (estimated to be 1100 days) such that the boric acid corrosion could reduce the surrounding carbon steel material below the ASME required wall. This potential issue is managed by periodic inspections every outage for evidence of leakage. These inspections are performed in accordance with OPs 1- and 2-0120022, and consist of an inspection during the initial plant cooldown from NOP conditions, a cold shutdown inspection, and a final inspection at NOP conditions during the plant startup.

There are eight additional nozzles in the hot legs that utilize Alloy 600 heat NX-0003 in the Unit 1 hot leg (A hot leg; RC-132, 133, 134, 135, 143 and in B hot leg; RC-127, 128, and 129). Since the other eight nozzles are in the hot leg with the same time at temperature, they are near the end of their predicted life. However, immediate replacement is not required since the failure mechanism in these small-bore nozzles is boric acid corrosion of the carbon steel piping. This effect is managed by the boric acid walkdowns every outage using plant procedure OP-1-0120022.

Heat NX-0003 is not used in any other location at St. Lucie Unit 1 or 2 other than those identified above.

Primary water stress corrosion cracking requires the presence of a susceptible material condition in Alloy 600 material, an aggressive environment (i.e. temperature, pure primary water), and tensile stress greater than some threshold value (residual, applied, or combination). Materials with the lowest susceptibility are characterized as having increased quantities of grain boundary (intergranular) carbides and relatively coarse (large) grains. Although it is difficult to determine the relative susceptibility of a specific material, it is generally expected that all Alloy 600 of the same heat (material chemistry and thermal processing) will perform similarly under identical conditions. PWSCC is also a thermally activated process that follows an Arrhenius relationship where an increase in temperature results in a decrease in time to initiation of cracking or failure. This temperature dependence is observed in Alloy 600 steam generator tube data where PWSCC was first observed in the tube sheet on the hot leg side followed several years later on the cold leg side.

To date, most cracking has been found in the Alloy 600 material adjacent to the partial penetration nozzle weld. This area is subjected to the shrinkage stresses from the weld and promotes predominately axial cracking. Field experience with PWSCC

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in these small diameter instrument nozzles has been that leakage is small and results in almost no boric acid corrosion of the surrounding carbon steel material. This is due to the effectiveness of the boric acid inspection programs (resulting from NRC Generic Letter (GL) 88-05) and the focus of these inspections on Alloy 600 locations. Since these programs have been implemented industry wide, although PWSCC has been detected, it has been detected before any boric acid corrosion could occur that would result in a safety concern.

Corrective Actions

1. RCS nozzle RC-126 was replaced under ASME Section XI IWA-7000 with a half nozzle design prior to startup.
2. The remaining eight 3/4-inch nozzles of Alloy 600 heat NX-0003 in the St. Lucie Unit 1 hot leg were visually inspected. No other failures were identified.
3. The Unit 1 and 2 inspection procedures (OPs 1- and 2-0120022, "Reactor Coolant System Leak Test") were reviewed and are effective at identifying the leakage associated with the small-bore PWSCC failures and no changes were required.
4. There are eight additional 3/4-inch nozzles of Alloy 600 heat NX-0003 in the St. Lucie Unit 1 hot legs. FPL is developing a replacement schedule for the remaining eight nozzles.

Additional Information

Failed Components Identified

Component: 3/4-inch RCS instrument nozzle

Material: Alloy 600

Manufacturer: ABB CE

Heat Number: NX-0003

Similar Events

LER 389/95-004 - Reactor Coolant System Instrument Nozzle Leakage Caused by Primary Water Stress Corrosion Cracking.

LER 3899/93-004 - This report describes pressurizer nozzle leakage due to primary stress corrosion cracking.

LER 389/94-002 - This report describes pressurizer instrument nozzle weld cracking due to fabrication defects in combination with primary water stress corrosion cracking.