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FACIL:50-389 St. Lucie Plant, Unit 2, Florida Power & Light Co. 05000389
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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 95-004-00:on 951010,RCS instrument nozzle leakage occurred.Caused by primary water stress corrosion cracking. Completed visual insp of other RCS instrument nozzles. W/951030 ltr.

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Florida Power & Light Company, P.O. Box 128, Fort Pierce, FL 34954-0128

FPL

October 30, 1995

L-95-292
10 CFR 50.73

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

Re: St. Lucie Unit 2
Docket No. 50-389
Reportable Event: 95-004
Date of Event: October 10, 1995
Reactor Coolant System Instrument Nozzle Leakage Caused by Primary Water Stress Corrosion Cracking

The attached Licensee Event Report is being submitted pursuant to the requirements of 10 CFR 50.73 to provide notification of the subject event.

Very truly yours,

D. A. Sager
Vice President
St. Lucie Plant

DAS/EJB

Attachment

cc: Stewart D. Ebneter, Regional Administrator, USNRC Region II
Senior Resident Inspector, USNRC, St. Lucie Plant

IE22

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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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
10	10	95	95 --	004 --	0	10	30	95	N/A	
									N/A	

OPERATING MODE (9)	3
POWER LEVEL (10)	0

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

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Edwin J. Benken, Licensing Engineer	(407) - 467 - 7156

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
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 October 10, 1995, St. Lucie Unit 2 was in Mode 3 following a shutdown for a scheduled refueling outage. A routine Reactor Coolant System (RCS) visual leak check was being performed in accordance with an approved plant procedure. During the course of the inspection, a utility Quality Control inspector observed that an instrument nozzle located on the 'B' side RCS hot leg exhibited an apparent boric acid buildup indicative of RCS leakage.

Although no active leakage was observed, a plant cooldown was continued to Mode 5 within the Technical Specification (TS) time requirements for RCS pressure boundary leakage. Further investigation confirmed that pressure boundary leakage had previously occurred , most probably due to primary water stress corrosion cracking (PWSCC) of alloy 600 material at the instrument nozzle.

The corrective actions for this event were: 1) A visual inspection of other RCS instrument nozzles was completed and no additional leakage was found. 2) An Isotopic analysis of the residue was performed which indicated that no recent RCS leakage had occurred, but that leakage had occurred at some time in the past. 3) Plant staff will conduct additional testing during nozzle removal to confirm the presence and orientation of the indication. 4) The defective instrument nozzle and other RCS nozzles having the same heat number will be replaced prior to unit startup. 5) Engineering will review the data collected from this inspection for enhancements to the existing RCS nozzle inspection program and schedule for alloy 600 nozzle replacement.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

DESCRIPTION OF THE EVENT

At approximately 1300 on October 10, 1995, St. Lucie Unit 2 was in mode 3 following shut down for a scheduled refueling outage. The Reactor Coolant System (RCS) (EIIS:AB) temperature was approximately 465^{0}F and primary pressure was approximately 1640 psia. The RCS was being visually inspected for leakage in accordance with an approved plant procedure. A utility Quality Control Inspector performing the inspection observed an apparent boric acid buildup at a Steam Generator (SG) (EIIS:AB) differential pressure instrument nozzle (PDT 1121B) (EIIS:AB) located on the 'B' RCS hot leg. No active leakage was observed at the time, however it was evident from the presence of boric acid residue at the nozzle that RCS leakage had previously occurred.

Operations was notified of the evidence that RCS pressure boundary leakage had occurred, and as a precautionary measure, the action requirements of Technical Specification (TS) 3.4.6.2 for pressure boundary leakage were implemented. TS 3.4.6.2 requires that with any pressure boundary leakage present, the plant must be in at least Hot Standby (Mode 3) within 6 hours and in Cold Shutdown (Mode 5) within the following 30 hours. The plant cooldown and depressurization was continued and Mode 5 was entered at 1438 on October 11, 1995.

CAUSE OF THE EVENT

The probable root cause of the instrument nozzle leakage indication is primary water stress corrosion cracking (PWSCC). Inconel 600 nozzles have proven to be susceptible to PWSCC as evidenced by similar industry events at San Onofre, Calvert Cliffs and St. Lucie plants. St. Lucie Engineering previously identified the heat number of the alloy 600 material used in this hot leg instrument nozzle as a susceptible heat based on industry failure history. Plant staff will conduct testing to verify the presence and orientation of the indication during nozzle removal, and attempt to confirm that PWSCC was the failure mechanism involved in this event.

ANALYSIS OF THE EVENT

The Chemistry Department performed an isotopic analysis of the residue found at the nozzle. This analysis revealed the presence of Cobalt 60 (Co-60), however no Cobalt 58 (Co-58) was detected. Since Co-60 has a half life of 5.27 years, and Co-58 has a half life of 70.8 days , it was concluded that no active leakage had occurred for a significant amount of time.

The RCS leakage path was apparently through a crack in the nozzle material near the internal attachment weld which is considered part of the RCS pressure boundary. As described in plant Technical Specifications, this is considered pressure boundary leakage. Although no active leakage was present at the time of discovery, it was conservatively assumed that pressure boundary leakage had existed during recent operations at 100 percent power. Therefore, this event is reportable under 10 CFR 50.73.a.2.i.B as operation prohibited by technical specifications.

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ANALYSIS OF THE EVENT Continued

The RCS hot leg instrument nozzle with the observed defect was associated with Steam Generator Differential Pressure Instrument (PDT) 1121B (EIIS:AB). This instrument measures reactor coolant flow and provides input logic to the Reactor Protective System (RPS) (EIIS:JC) for tripping the reactor on a low flow condition. The operability of PDT-1121B during normal operation and cooldown would not have been affected, since leakage from instrument nozzles resulting from PWSCC is expected to be very small. Analysis has shown that leakage resulting from PWSCC will be confined to slowly increasing leak rates in the fraction of a gallon per minute range.

St. Lucie Engineering evaluation JPN-PSL-SEMP-93-013, Revision 1, "Cracking of Reactor Coolant System Instrument Nozzles and Pressurizer Heater Sleeves" performed in March of 1993, evaluated pressurizer instrument nozzle cracking due to PWSCC and the significance to plant operation and safety. In this evaluation, Combustion Engineering Owners Group (CEOG) program findings were used to assess the potential for catastrophic failure of the nozzles by either circumferential or axial rupture. The evaluation examined various failure modes and a predictive analysis methodology was performed for the nozzles. The evaluation concluded that circumferential cracking leading to catastrophic failure of an instrument nozzle from a guillotine or slot break was not a credible failure mode. Additionally, it was determined that stress corrosion cracks in instrument nozzles would be axial and confined to areas near the J-groove partial penetration welds. The average leak rate from a crack as measured in tests and reported in the above evaluation was in the range of .014 to .11 gpm. Predictive analysis to determine the remaining life or time to failure for the installed nozzles concluded that the instrument nozzles susceptible to PWSCC would not affect the safe operation of St. Lucie 1 and 2.

The primary water stress corrosion cracking of alloy 600 has been analyzed as an industry condition and is a condition which also affects St. Lucie units 1 and 2. Based on the Engineering evaluations performed at St. Lucie , the as found leakage condition of the instrument nozzle described in this event is an analyzed condition, and has been determined to not seriously degrade plant safety barriers or compromise safe plant operation. The health and safety of the public was not affected by this event.

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CORRECTIVE ACTIONS

1. A visual inspection was performed of other hot leg and pressurizer nozzles and no other evidence of leakage was found.

2. Chemistry performed an isotopic analysis of the residue found at the affected instrument nozzle which confirmed that no recent RCS leakage had occurred, but that leakage had occurred at some time in the past.

3. Plant staff will conduct testing to verify the presence and orientation of the indication during nozzle removal and attempt to confirm that primary water stress corrosion cracking (PWSCC) was the failure mechanism in this event.

4. The defective instrument nozzle and other instrument nozzles which are of the same heat number as the leaking nozzle in this event, will be replaced with alloy 690 material prior to Unit 2 startup.

5. Engineering will review the data resulting from this event and update the existing nozzle susceptibility matrix for both St. Lucie Units 1 and 2.

ADDITIONAL INFORMATION

Failed Component

Component: Instrument Nozzle on Reactor Coolant System
Material: Iconel 600
Manufacturer: ABB CE
Heat Number: NX 7630

Previous Similar Events

LER 389/93-004 - This report describes Pressurizer instrument nozzle leakage due to primary water stress corrosion cracking (PWSCC) involving alloy 600 material.

LER 389/94-002 - This report describes Pressurizer instrument nozzle weld cracking due to fabrication defects in combination with primary water stress corrosion cracking (PWSCC) of alloy 600 equivalent material.