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

FPL

May 12, 2005

L-2005-108
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: 2005-001-00
Date of Event: March 17, 2005
Operation With Inoperable Steam Generator Water Level Channel

The attached Licensee Event Report 2005-001 is being submitted pursuant to the requirements of 10 CFR § 50.73 to provide notification of the subject event.

Very truly yours,

William Jefferson, Jr.
Vice President
St. Lucie Nuclear Plant

WJ/KWF

Attachment

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE

Operation With Inoperable Steam Generator Water Level Channel

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
03	17	2005	2005	- 001 -	00	05	12	2005	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Kenneth W. Frehafer, Licensing Engineer	(772) 467 - 7748

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	SB	V	D232	NO	-	-	-	-	-

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	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)*

On March 17, 2005, St. Lucie Unit 1 was in Mode 1 at 100 percent reactor power. Maintenance personnel initiated a condition report to troubleshoot a 3 to 4 percent high deviation between the 1B steam generator narrow range water level indicator, LIC-9023A, and its three redundant level channels. The subsequent Engineering evaluation concluded that steam generator level channel indication deviations greater than 2 percent were outside the accuracy assumed in the analysis, and Operations declared the channel out of service on March 22, 2005.

A leak in the reference leg caused a false high indicated water level. The leak was repaired, and on March 25, 2005, the channel was declared back in service. Additional corrective actions include enhanced procedural guidance for steam generator water level cross-channel check acceptance criteria.

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

Description of the Event

On March 17, 2005, St. Lucie Unit 1 was in Mode 1 at 100 percent reactor power. Maintenance personnel were investigating an existing work order concerning a 3 to 4 percent high deviation between the 1B steam generator narrow range water level indicator, LIC-9023A [EIIS:SB:LIT], and its three redundant level channels. Maintenance narrowed the deviation cause to a possible leak through the 5-way manifold in containment, a small external leak on the reference leg, and/or transmitter calibration drift issues. Maintenance initiated a condition report to further troubleshoot the condition. The subsequent Engineering evaluation concluded that steam generator level channel indication deviations greater than 2 percent were outside the accuracy assumed in the analysis, and Operations declared the channel out of service on March 22, 2005. The cause was diagnosed as a reference leg problem in the water level instrument, LT-9023A [EIIS:SB:LT]. On March 25, 2005, the channel was declared back in service after repairs were complete.

Cause of the Event

The investigation into this event discovered a leaking instrument valve located within the reference leg boundary. A valve [EIIS:SB:V] in the upper sensing line attached to the reference leg condensing pot had a packing leak. The instrument valve packing leakage exceeded the make-up ability of the condensing pot and voided a portion of the reference leg. The lowered reference leg water level caused the channel to indicate a higher water level than actual.

On March 25, 2005, the valve packing leak was corrected and the reference leg was back filled to restore the level in the condensing pot. These corrective actions restored the level indication of LIC-9023A to within 1 percent of the other three steam generator level channels.

Additionally, although a work order was initiated to investigate the deviation, the control board operators did not report the variances in the 1B steam generator levels as an operability concern due to inadequate channel check acceptance criteria. Data sheet 26, "Accident Monitoring Instrumentation," of each unit's procedure OP-0010125A, "Surveillance Data Sheets," was revised include steam generator level channel check acceptance criteria.

Analysis of the Event

This event is reportable under 10 CFR 50.73(a)(2)(i)(B) as operation or a condition prohibited by the Technical Specifications. The inoperable condition exceeded the allowed outage times for an inoperable steam generator water level indicator as specified in Technical Specifications 3/4.3.1, "Reactor Protective Instrumentation," and 3/4.3.2, "ESFAS Instrumentation." A review of the past operation, using Operator shiftly channel checks and available instrument data, indicated a notable adverse trend in the steam generator water level measured by LT-9023A as early as January 2005. The condition worsened until, on February 19, 2005, Operations initiated a work order (WO) to investigate the noted 3 to 4 percent deviation.

As stated in the Technical Specifications, "A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation." A numerical acceptance criteria is not required. However, due to the fact that Technical Specification operability could be called into question for small deviations in indicated steam generator level, additional Engineering guidance was provided. Although the pressurizer level transmitters have similar designs, there is no

Technical Specification operability question associated with a small drift in pressurizer level. FPL is evaluating the acceptability of the existing acceptance criteria for all remaining Technical Specification instrument channel checks.

Analysis of Safety Significance

The indicated 1B steam generator level was 3 to 4 percent higher than the other channels. LT-9023A is one of four 1B steam generator level signals and feeds the Unit 1 "A" channel reactor protection system (RPS) and auxiliary feedwater actuation system (AFAS). Although the "A" channel was inoperable, it was still capable of providing an RPS trip and AFAS initiation on a low level signal, however, the signal associated with the "A" channel would lag behind with respect to the other three channels. In addition, the other three channels remained operable and any two of these three would have been capable of providing the necessary RPS and/or AFAS actuations.

The events where the signal delay could potentially have an adverse effect on the analysis results were evaluated. These events were the loss of feedwater event, loss of offsite power event, station blackout event, small break LOCA event, and the Updated Final Safety Analysis Report (UFSAR) auxiliary feedwater system evaluation. The results of this analysis concluded that the UFSAR analyses acceptance criteria were met with a 3 to 4 percent delay in the "A" channel RPS and AFAS signals.

Based on the above, this event had no adverse impact on the health and safety of the public.

Corrective Actions

1. The reference leg leak was repaired under WO 35003748.
2. Both units revised Data sheet 26, "Accident Monitoring Instrumentation," of the unit specific procedure OP-0010125A, "Surveillance Data Sheets," to include steam generator level channel check acceptance criteria.
3. The evaluation of all remaining Technical Specification instrument channel check acceptance criteria will be complete by July 15, 2005.

Other Information

Failed Equipment Identified

Component: 1/2" 5-Way Manifold Valve

Manufacturer: Dragon Valve Inc.

Model Number: 13914N-8SE

Similar Events

None