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

FPL

October 16, 2007

L-2007-163
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: 2007-001-00
Date of Event: August 18, 2007
Reactor Shutdown Due to Unidentified RCS Leakage

The attached Licensee Event Report 2007-001-00 is being submitted pursuant to the requirements of 10 CFR 50.73 to provide notification of the subject event.

Very truly yours,

Gordon L. Johnston
Site Vice President
St. Lucie Plant

GLJ/dlc

Attachment

IE22
NRR

NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER)		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.

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4. TITLE

Reactor Shutdown Due to Unidentified RCS Leakage

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	18	2007	2007	- 001 -	00	10	16	2007	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)
Donald L. Cecchett - Licensing Engineer	772-467-7155

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	AB	PSF	N/A	YES					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO		-	-	-

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On August 18, 2007, St. Lucie Unit 2 was in Mode 1 at 100 percent reactor power when a planned manual reactor trip was initiated. The planned trip was a result of a known reactor coolant system (RCS) unidentified leak exceeding a plant management imposed administrative leakage limit. Technical Specification operational leakage limits were never exceeded. The subsequent containment walk down identified the leak source as non-isolable RCS pressure boundary leakage from a cracked socket weld on the 3/4 inch Class 1 2B1 Reactor Coolant Pump (RCP) seal injection piping. The cause for the socket weld failure was low stress high cycle fatigue resulting from RCP vibration and the susceptibility of socket welds to high cycle fatigue failure.

During the forced outage, the failed weld was removed and the seal injection piping was isolated, with the configuration controlled under a temporary system alteration. Corrective actions include replacement of the most vulnerable Unit 2 RCP seal injection piping socket welds with a more robust weld design, permanent removal of insulation from sockets welds within the Unit 2 RCP shrouds to facilitate future inspections, replacement of the 2B1 RCP motor, and evaluation of the St. Lucie Unit 1 seal injection piping.

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

Description of the Event

On February 20, 2007, reactor coolant system (RCS) unidentified leakage increased from 0.04 gpm to 0.14 gpm over the course of a week. Elevated reactor cavity level and sump pump run times coincided with the step change in unidentified leakage. A minor increase in containment radiation levels was also observed. Station personnel mobilized an Event Response Team (ERT) to locate and evaluate the source of the leakage. All accessible areas were inspected by walkdowns which found no active leakage.

Remote inspection was attempted on three occasions, first using a robot owned and operated by Florida Power & Light (FPL). No active leakage was found during the first inspection. The second remote inspection (February 2007) was performed using a vendor supplied robot. A small amount of water was observed under Reactor Coolant Pump (RCP) [EIIS:AB:P] 2B1. The third remote inspection (August 2007) was performed using a robot with extension capabilities which found a puddle of water under the RCP 2B1 which was brown in color (rust). However, the robot failed as it was being deployed to observe the RCP seal package and was unable to identify the source of the water. The leakage increased to a previously established administrative action limit which resulted in a manual shutdown of Unit 2 on August 18, 2007. Following the shutdown a containment walk down was performed; the leak was identified as non-isolable RCS pressure boundary leakage from a cracked socket weld [EIIS:AB:PSF] on the 3/4 inch Class 1 2B1 RCP seal injection piping.

During the forced outage the failed weld was removed and the seal injection piping was isolated, with the configuration controlled under a temporary system alteration (TSA). The TSA will be removed and the seal injection piping restored during the fall 2007 SL2-17 refueling outage.

Cause of the Event

The cause of the socket weld failure was low stress high cycle fatigue. Resonance excitation of the seal piping resulted in cyclic loading at the weld and propagation of the crack to failure. At the time the seal piping was installed, the small bore piping design process did not specifically require evaluation for resonance conditions. Additional contributing causes were the inherent susceptibility of the socket weld to fatigue cracking, when subjected to cyclic loading conditions, and the excitation of the seal piping caused by the 2B1 RCP vibration. The 2B1 RCP has higher vibration compared to the other St. Lucie Unit 2 RCPs.

Analysis of the Event

This event is reportable under 10 CFR 50.73(a)(2)(i)(B) as operation or condition prohibited by the Technical Specifications (TSs). TS 3/4.4.6.2, Operational Leakage, does not allow operation with pressure boundary leakage. TS 3/4.4.11, Structural Integrity, requires maintaining the structural integrity of ASME Code Class 1, 2, and 3 components. Contrary to these TS requirements, St Lucie Unit 2 operated for approximately 6 months with RCS pressure boundary leakage. Additionally, this condition 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.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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Analysis of Safety Significance

Plant operation was within the TS operational limits for unidentified leakage, and the unidentified operational leakage limit of 1 gpm considers that the potential source of the unidentified leakage may be pressure boundary leakage. Therefore, there was no reduction in the margin of safety and no adverse impact on the health and safety of the public.

An assessment of a postulated break of the seal injection piping concluded that the worst-case leak rate would be well within the makeup capacity of one charging pump; water injection from the emergency core cooling system (ECCS) would not be required to mitigate the break. The condition would remain bounded by the small break loss of coolant analyses performed as part of the ECCS performance analysis. Therefore, the rupture of the seal injection piping would not have resulted in any unanalyzed condition which would challenge any safety analysis criteria.

The risk significance of this condition is based upon the Conditional Core Damage Probability (CCDP) of the event in question. The calculated CCDP determines the increase in Core Damage Frequency (CDF) incurred by the configuration and the duration the unit was in this configuration. The only effect of this event on CDF would be a manual shutdown due to excessive unidentified leakage. As stated above, worst-case leakage would be within the makeup capability of the charging pumps so the probabilistic model treats this condition as a general reactor trip. The event lasted 186 days which resulted in a CCDP of approximately 2.4E-7, which is considered low risk.

RCP seal injection flow is provided from the chemical and volume control system (CVCS) and is a backup to the component cooling water (CCW) RCP seal cooling. Since RCP seal injection flow is only used during RCS fill and vent evolutions following extended shutdowns, the loss of seal injection flow to the 2B1 RCP would have no adverse impact on plant operation or safe shutdown capability.

Failure analysis performed on the cracked weld concluded the crack was comprised of multiple initiation points at the weld root, with subsequent propagation through the throat of the weld. The failure mechanism was low stress high cycle fatigue caused by vibration induced cyclic loading.

Although stainless steel components are not susceptible to boric acid corrosion, the spray from the cracked weld deposited wet boric acid onto the external surfaces of numerous carbon and low alloy steel components, including pressure retaining bolting. Inspection of the affected areas revealed only minor surface corrosion with no significant material degradation or wastage. The inspection results were consistent with the expected corrosion rates provided in the industry guidelines for the specific leak parameters such as temperature and boric acid concentration.

Based on the above, the safety consequences of the event are judged to be low and there was no adverse impact on the health and safety of the public.

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Corrective Actions

The proposed corrective actions and supporting actions listed below will be entered into the site corrective action program. Any changes to the proposed actions will be managed under the commitment management change program.

1. Replace seal injection piping on all four Unit 2 RCPs, using a more robust weld design rather than socket welds in the most vulnerable locations.
2. Permanently remove the insulation from the seal piping inside the Unit 2 RCP shrouds to facilitate inspection and early detection of any future weld cracks.
3. Replace 2B1 RCP motor.
4. Evaluate Unit 1 RCP piping to upper, middle, and lower seal cavities to determine whether the actions taken on Unit 2 are applicable to Unit 1.
5. Evaluate this event and incorporate lessons learned into applicable engineering procedures and instructions to address vibration and high cycle fatigue and use of socket weld fittings.

Actions 1, 2, and 3 above will be completed prior to the end of the fall 2007 Unit 2 refueling outage.

Similar Events

1. LER-335-87014, PSL Unit 1, 10-8-87, cracked weld at lower cavity seal nozzle flange. Cracked weld was attributed to a combination of relatively high RCP vibration and flange misalignment between the nozzle and injection piping.
2. Condition Report 2005-4621 documents the failure of a 3/4 inch socket weld on the 2B1 safety injection piping. This failure was due to a defective weld along with high cycle fatigue.

Failed Components

3/4 inch Schedule 160 piping socket weld (RC-227/SW-2) to 45 degree elbow