Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.73

November 19, 2003
3F1103-03

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: LICENSEE EVENT REPORT 50-302/03-003-00

Dear Sir:

Please find enclosed Licensee Event Report (LER) 50-302/03-003-00. The LER discusses small Reactor Coolant System pressure boundary leaks found in three upper level pressurizer instrument nozzles. This report is being submitted pursuant to 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(ii)(A).

No new regulatory commitments are made in this letter.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

JH Terry

James H. Terry
Manager, Engineering
Crystal River Nuclear Plant

JHT/dwh

Enclosure

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

IE22

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
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 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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 Coolant System Pressure Boundary Leakage Limit Exceeded Due To Pressurizer Instrument Tap Nozzle Cracks

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
10	04	2003	03	- 003	- 00	11	19	2003	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	4
10. POWER LEVEL	0%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	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(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(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)	X	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)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Dennis W. Herrin, Lead Engineer	(352) 563-4633

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	AB	NZL	B015	Y					

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

At 1032, on October 4, 2003, Progress Energy Florida, Inc., (PEF's) Crystal River Unit 3 (CR-3) was in MODE 4 (HOT SHUTDOWN) at zero percent RATED THERMAL POWER. While performing a visual inspection of the pressurizer upper level instrument tap nozzles, evidence of a very small Reactor Coolant System (RCS) pressure boundary leak was found at three nozzles. Improved Technical Specification (ITS) 3.4.12 states that RCS operational leakage shall be limited to no pressure boundary leakage in MODES 1, 2, 3, or 4. If pressure boundary leakage exists, the plant shall be in MODE 3 within 6 hours and in MODE 5 within 36 hours. At 1741 on October 4, 2003, CR-3 entered MODE 5. The identified condition existed during plant operation and is being reported under 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(ii)(A). This condition does not represent a reduction in the public health and safety. The most probable cause for this condition was primary water stress corrosion cracking (PWSCC). The three pressurizer upper level instrument tap nozzles were repaired. This is the first occurrence of pressurizer level instrument tap nozzle pressure boundary leakage at CR-3.

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

BACKGROUND

In late 2000, an increase in Reactor Building (RB) [NH] airborne activity was noted when pressurizer spray was initiated. During Refueling Outage 12 in October 2001, the pressurizer spray valve [AB, V] was repacked and the insulated pressurizer spray line [AB, PZR] received a focused walk down, in addition to the routine boric acid and Reactor Coolant System (RCS) leakage walk downs. After returning to power operations, the RB airborne activity still increased when spray was initiated. A team was formed to address the multiple concerns raised by this phenomenon. The team's activities are documented in the CR-3 Corrective Action Program under Nuclear Condition Report (NCR) 69214. Data gathered on-line, walk downs during forced outages, and external operating experiences led to identifying the pressurizer steam space as the most likely source of the radionuclides escaping into the RB. Therefore, the most likely leak sites were nozzles, flanges or piping directly connected to the pressurizer steam space.

Focused inspection teams were formed to inspect selected locations on the pressurizer and connected piping during Refueling Outage 13. On October 4, 2003, initial inspections were performed in MODE 3 because the higher plant pressure would facilitate leak detection. Ultrasonic guns were used to enhance the ability to acoustically detect a leak. During the walk downs, ultrasonic noise was detected at the joint between the RC-1-LT3 upper level instrument tap nozzle and the pressurizer shell. Similar noise was detected in the vicinity of the upper level instrument tap nozzle for RC-1-LT1. After plant conditions allowed insulation removal, a visual inspection was performed.

EVENT DESCRIPTION

At 1032, on October 4, 2003, Progress Energy Florida, Inc., (PEF's) Crystal River Unit 3 (CR-3) was in MODE 4 (HOT SHUTDOWN) at zero percent RATED THERMAL POWER. While performing a visual inspection of pressurizer upper level instrument tap nozzles RC-1-LT1, RC-1-LT2 and RC-1-LT3 [AB, NZL], evidence of a very small RCS pressure boundary leak was found at each nozzle. The leakage evidence consisted of light boric acid deposits and stains on the pressurizer carbon steel shell for the RC-1-LT1 and RC-1-LT3 nozzles. Stains, but no boric acid deposits, were found on the pressurizer carbon steel shell for the RC-1-LT2 nozzle. No leakage evidence was found on any of the similar nozzles: the lower level taps, resistance temperature detector (RTD) thermowell, water sample tap or vent nozzle. The inspection was performed to determine the source of increased RB airborne activity noted when pressurizer spray was initiated.

Improved Technical Specification (ITS) 3.4.12 states that RCS operational leakage shall be limited to no pressure boundary leakage in MODES 1, 2, 3, or 4. If pressure boundary leakage exists, the plant shall be in MODE 3 within 6 hours (Action B.1) and in MODE 5 within 36 hours (Action B.2). At 1741 on October 4, 2003, CR-3 entered MODE 5.

At 1611, on October 4, 2003, this condition was reported as a non-emergency eight-hour notification to the NRC Operations Center (NRC Event 40222) under 10CFR50.72(b)(3)(ii)(A).

The identified condition existed during plant operation and is being reported under 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(ii)(A).

SAFETY CONSEQUENCES

The pressurizer [AB, PZR] is an integral part of the RCS pressure boundary. The pressurizer serves the following functions: (a) establishes and maintains RCS pressure; (b) provides a surge volume for limiting pressure excursions; and, (c) provides a water reserve to accommodate reactor coolant temperature/density changes during operation. The primary safety function provided by the pressurizer is to maintain RCS pressure boundary integrity.

Industry operating experience with pressurizer nozzle cracks dates to the 1980's. Experience shows that periodic visual inspection detects the cracks before they grow large enough to challenge the structural stability of the nozzle. In all cases, when metallurgical failure analyses were performed, the cracking was axial in direction. There is no evidence that a pressurizer instrument penetration has ever developed a significant circumferential flaw. The RCS unidentified leakages never approached the 1 gallon per minute (gpm) Technical Specification shutdown limit. Since failure analyses have indicated that cracking is in the axial direction, there is minimal risk of a circumferential break of the nozzle. Relatively low axial stresses, in comparison with radial stresses, in these small nozzles significantly reduce the possibility of developing circumferential cracks, even after leakage has developed.

In the case at CR-3, the RCS unidentified leakage rate was very small and never approached the 1 gpm Technical Specification limit. There was no appreciable accumulation of boric acid or wastage of the carbon steel shell, which was confirmed by visual inspection of the pressurizer shell around the three leaking nozzle penetrations and ultrasonic testing of the base material. This is consistent with Operating Experience. Only staining and light boric acid residue indicated the presence of a leak. Therefore the event had low safety significance.

Based on the above discussion, this condition does not constitute a Safety System Functional Failure as defined in Nuclear Energy Institute document 99-02 and there was no reduction in the public health and safety.

CAUSE

Based upon the review of the facts, material of construction, fabrication methods, characteristics of the leakage detected and Operating Experience, CR-3 concludes that primary water stress corrosion cracking (PWSCC) is the most likely failure mechanism for the pressurizer upper level instrument tap nozzle cracking. The use of Alloy 600/82/182 led to the development of PWSCC, which eventually grew to provide a path through which the reactor coolant was able to leak to the

outside of the pressurizer. This failure mode is consistent with the PWSCC failures in RCS nozzles of similar material, design, installation practice, and environment.

The Alloy 600/82/182 material is susceptible to PWSCC. High residual stresses created from the welding process, the environment created by the RCS water chemistry, and operating temperatures promote PWSCC. The amount of leakage observed is consistent with leakage seen at other Pressurized Water Reactors in the same general location (steam space).

CORRECTIVE ACTIONS

1. The three pressurizer upper level instrument tap nozzles were repaired using the "half-nozzle" technique (Engineering Change 51801 and Work Order 469645). The technique leaves the flaw in place, but moves the pressure boundary from the internal weld to an external location. Additionally, it replaces a segment of the Alloy 600 nozzle with Alloy 690, using a similar weld material (Alloy 52 or 52M), which are less susceptible to PWSCC at operating temperatures.

2. A review of Operating Experience suggested that leakage would be associated with the pressurizer steam space versus the water space. This was substantiated by no evidence of RCS leakage being found during bare metal inspections on any of the similar pressurizer nozzles: the lower level taps, RTD thermowell, water sample tap or vent nozzle.

3. Other actions (e.g., evaluating the need to revise the augmented Inservice Inspection program to bare metal inspect the pressurizer vent, lower level instrument taps, RTD thermowell, and water sample nozzle on some frequency) are being addressed in NCR 106443.

PREVIOUS SIMILAR EVENTS

This is the first occurrence of pressure boundary leakage of a pressurizer level instrument tap nozzle at CR-3. However, an occurrence of RCS pressure boundary leakage due to PWSCC of Alloy 600 material was reported to the NRC in Licensee Event Report 50-302/01-004-00, "Reactor Pressure Vessel Head Leakage Due To Control Rod Drive Mechanism (CRDM) Nozzle Degradation," dated November 28, 2001.

ATTACHMENTS

Attachment 1 - Abbreviations, Definitions, and Acronyms

Attachment 2 - List of Commitments

LICENSEE EVENT REPORT (LER)

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

ATTACHMENT 1

ABBREVIATIONS, DEFINITIONS AND ACRONYMS

CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
gpm	gallons per minute
ITS	Improved Technical Specifications
NCR	Nuclear Condition Report
PEF	Progress Energy Florida, Inc.
PWSCC	Primary Water Stress Corrosion Cracking
RB	Reactor Building
RCS	Reactor Coolant System
RTD	Resistance Temperature Detector

NOTES:

Improved Technical Specifications defined terms appear capitalized in LER text {e.g., MODE 1}

Defined terms/acronyms/abbreviations appear in parenthesis when first used {e.g., Reactor Building (RB)}.

EIIS codes appear in square brackets {e.g., reactor building penetration [NH, PEN]}.

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

ATTACHMENT 2

LIST OF COMMITMENTS

The following table identifies those actions committed to by Progress Energy Florida, Inc., (PEF) in this document. Any other actions discussed in the submittal represent intended or planned actions by PEF. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing & Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

RESPONSE SECTION	COMMITMENT	DUE DATE
	No regulatory commitments are being made in this submittal.	