Progress Energy

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

Ref: 10 CFR 50.73

November 22, 2004
3F1104-03

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

Subject: LICENSEE EVENT REPORT 50-302/2004-004-00

Dear Sir:

Please find enclosed Licensee Event Report (LER) 50-302/2004-004-00. The LER discusses a Notice of Clarification issued by the NRC regarding Steam Generator Tube Integrity Event Reporting Guidelines for NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 10 CFR 50.73," which directed licensees to consider the results of the previous steam generator tube inspections against the NUREG-1022 revised guidelines. Crystal River Unit 3 determined that the as-found steam generator projected leakage value for Steam Line Break, which exceeded the leak rate limit for Refueling Outage 13 (13R), is reportable per the revised NUREG-1022 guidance. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(A).

No 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,

Jon A. Franke
Plant General Manager
Crystal River Nuclear Plant

JAF/lvc

Enclosure

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

IE22

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

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.

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4. TITLE
NUREG-1022 Clarification Required Reporting of Previous Steam Generator Tube Inspection Results

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: N/A	DOCKET NUMBER: 05000
09	24	2004	2004	- 004 -	00	11	22	2004	FACILITY NAME: N/A	DOCKET NUMBER: 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

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

☐ 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)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 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

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Loretta V. Cecilia– Senior Engineer (Licensing & Regulatory Programs)	(352) 563-4546

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
E	AB	SG	B015	Y					

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)*

At 14:00, on September 24, 2004, Progress Energy Florida, Inc., Crystal River Unit 3, was in MODE 1 (POWER OPERATION) at 100 percent RATED THERMAL POWER. The NRC published a Notice of Clarification to Steam Generator Tube Integrity Event Reporting Guidelines in NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 10 CFR 50.73." The Notice of Clarification revised the guidelines used to determine reporting of steam generator inspection results. The NRC directed licensees to consider the results of the previous steam generator tube inspections, including the structural integrity criteria and leakage criteria, against the new reporting guidelines. Crystal River Unit 3 determined that the as-found steam generator projected leakage value for Steam Line Break (SLB), which exceeded the leak rate limit for Refueling Outage 13 (13R), is reportable per the revised NUREG-1022 guidance, under 10 CFR 50.73(a)(2)(ii)(A), as a condition of the nuclear power plant, including its principal safety barriers, being seriously degraded. The cause for exceeding the leakage criterion was finding new indications not accounted for in the previous cycle Operational Assessment, and human error in the leakage calculation. Repairs performed during 13R reduced the SLB projected leakage to well below the limit prior to plant startup. This condition does not represent a reduction in the public health and safety. No similar occurrences have been previously reported.

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

EVENT DESCRIPTION

At 14:00, on September 24, 2004, Progress Energy Florida, Inc. (PEF), Crystal River Unit 3 (CR3), was in MODE 1 (POWER OPERATION) at 100 percent RATED THERMAL POWER. While reviewing NRC Federal Register Notice / Volume 69, Number 185, Pages 57367-57368, dated September 24, 2004, "Notice of Clarification to Steam Generator Tube Integrity Event Reporting Guidelines in NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 10 CFR 50.73," CR3 personnel became aware of the potential for reversing the reportability determination performed for a previously evaluated condition.

The NUREG-1022 clarification revised the guidelines used to determine whether the results of steam generator inspections were reportable. The previous NUREG-1022 sentence read: "Steam generator tube degradation is considered serious if the tubing fails to meet the following two performance criteria:..." The revised NUREG-1022 reads: "Steam generator tube degradation is considered serious if the tubing fails to meet either of the following two performance criteria:..." The Federal Register Notice directed licensees to consider the results of the previous steam generator tube inspections against the revised guidelines.

CR3 reviewed the results of the previous steam generator [AB, SG] inspection conducted during Refueling Outage 13 (13R) which occurred during October 2003. Under the revised NUREG-1022 guidelines, CR3 concluded that the as-found projected leakage for Steam Line Break (SLB during 13R was reportable. CR3 previously notified the NRC in Special Report 04-01, Results of the Once-Through Steam Generator Tube Inservice Inspection Conducted During Refueling Outage 13, that the as-found projected leakage for SLB exceeded the CR3 leak rate limit of 0.856 gallons per minute (gpm). The Condition Monitoring Assessment requirements for structural integrity were met during the previous operating cycle. Repairs performed during 13R inspection reduced the as-found SLB projected leak rate below the limit of 0.856 gpm prior to plant startup.

<u>Review of the Results of the Previous Steam Generator [SG] Inspection</u>

During 13R, CR3 performed the required Condition Monitoring and Operational Assessment (CMOA) evaluations. The condition of a steam generator, in terms of structural and leakage integrity, is determined by the nature of defects present in the steam generator Reactor Coolant [AB] Pressure Boundary. As such, the process of Condition Monitoring involves the evaluation of inspection results at the end of the operating interval to determine the state of the steam generator tubing for the most recent period of operation. The condition of the steam generator is then measured and the results compared against defined structural and leakage integrity performance criteria. This evaluation process is required by Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines."

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Condition Monitoring provides a means of verifying end-of-cycle tube integrity, checking previous operational assessment predictions, and identifying tubes requiring repair. If, upon completion of the evaluation, the results indicate that one or more tubes fail to satisfy the performance criteria specified in NEI 97-06, the condition should be documented. In addition, necessary corrective actions must be identified prior to restart. If Condition Monitoring indicates a significant difference between what is found in the steam generators and the operational assessment prediction from the previous outage, analyses must be performed to identify the reason for the difference.

Degradation other than Tube End Cracking (TEC) did not significantly challenge the required leakage integrity limits under postulated accident conditions. However, using conservative calculations of leakage approved by the NRC for TEC Alternate Repair Criteria (ARC) methodology led to a total Condition Monitoring SLB leak rate above the CR3 limit. The SLB leak rate limit for CMOA is 1 gpm minus the operational limit of 150 gallons per day (gpd), converted to SLB conditions for a limit of 0.856 gpm. The projected leakage based on the as-found condition for "A" Once-Through Steam Generator (A-OTSG) was 1.029 gpm and B-OTSG was 1.241 gpm.

Under the revised NUREG-1022 reporting guidelines, the above condition is considered to be reportable. At 17:35, on September 24, 2004, a non-emergency eight-hour notification was made to the NRC Operations Center (Event Number 41069) in accordance with 10 CFR 50.72(b)(3)(ii)(A). This report is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(A).

SAFETY CONSEQUENCES

The safety significance of exceeding the projected primary-to-secondary OTSG leakage is minor. TEC indications do not challenge the structural integrity of any tube because the degradation is contained within the tubesheets where burst can not occur. The postulated projected SLB leakage is calculated considering actual and potential defects in the steam generator tubing. The projected SLB leakage contains several very conservative assumptions about the TEC leak rates along with the assumption that every indication is 100% thru-wall and leaks. Compared to the actual OTSG primary-to-secondary leakage, the projected leakage for the previous operating cycle would have been over 40 gpd compared to an actual 2.5 gpd. The physical degradation of the tubing has been thoroughly evaluated by the nuclear industry and continues to be evaluated by utility Owners Groups.

Based on the above discussion, PEF concludes that the identified condition did not represent a reduction in the public health and safety. The OTSGs remained operable. Therefore, this event does not meet the NEI definition of a Safety System Functional Failure (NEI 99-02, Revision 2).

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

CAUSE

The inspection identified new TEC indications that were not accounted for in the previous cycle OA. CR3 used the TEC ARC probability of detection (POD) as the only margin for increased TEC leakage from one refueling outage to the next, and the calculation method for TEC growth rates (new indications) in Refueling Outage 12 (12R) was not included in plant procedures. This resulted in the as-left leakage margin being insufficient to account for the initiation of new TEC indications between the 12R and 13R inspection outages.

During 12R, CR3 selected an incorrect Leak Rate Table (reference CR3 to NRC letter dated May 28, 1999). This error also provided a small contribution for the TEC leakage calculation to be lower than it should have been. When discovered during 13R, using the corrected leak rates caused the postulated as-found leakage to be higher than expected.

CORRECTIVE ACTIONS

CR3 repaired the 13R as-found tubes with high TEC leak rates to reduce the 13R as-left TEC leakage and provide a margin for future new TEC.

The 12R TEC calculation 32-5014980-00, "CR-3 Tube End Cracking ARC Leakage Calculation – 12RFO," was revised using the correct Leak versus Radius table for CR3.

The system engineer classified the OTSG portion of the Reactor Coolant System as being (a)(1) in the Maintenance Rule.

The 13R TEC calculation was completed using the correct assumptions for leakage. The 13R calculation 32-5035732-00, "CR-3 RFO-13 TEC ARC Leakage Calculation," was reviewed and accepted by CR3 Engineering.

CR3 revised the Steam Generator Program Manual to include an expanded section to provide information on the CR3 specific TEC leakage tables to ensure the TEC calculations are performed correctly. This section identifies that the Operational Assessment must consider the past (historical) TEC growth and applicable leakage rates when predicting future cycle operation and not just rely on the TEC ARC application for POD.

Additional corrective actions are identified in the CR3 Corrective Action Program under Nuclear Condition Report 111677.

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PREVIOUS SIMILAR EVENTS

No previous similar events involving exceeding the as-found SLB projected leakage have been reported to the NRC by CR3.

ATTACHMENTS

Attachment 1 - Abbreviations, Definitions, and Acronyms

Attachment 2 - List of Commitments

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ATTACHMENT 1

ABBREVIATIONS, DEFINITIONS AND ACRONYMS

12R	Refueling Outage 12
13R	Refueling Outage 13
ARC	Alternate Repair Criteria
CFR	Code of Federal Regulations
CR3	Crystal River Unit 3
CMOA	Condition Monitoring and Operational Assessment
GPD	Gallons per Day
GPM	Gallons per Minute
SLB	Steam Line Break
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OTSG	Once Through Steam Generator
PEF	Progress Energy Florida, Inc.
POD	Probability of Detection
TEC	Tube End Cracks

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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ATTACHMENT 2

LIST OF COMMITMENTS

The following table identifies those actions committed to by 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.	