Progress Energy

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

Ref: 10 CFR 50.36

October 31, 2003
3F1003-07

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

Subject: Crystal River Unit 3 - Special Report 03-01: Once Through Steam Generator (OTSG) Notifications Required Prior to MODE 4

Reference: FPC to NRC Letter 3F0601-07, dated June 28, 2001, " Crystal River Unit 3 – Contingency Letter of Commitment Regarding License Amendment Request 252, Revision 0, Once Through Steam Generator Tube Surveillance Program, Tube Repair Roll (Re-Roll) Process" (TAC No. MB1519)

Dear Sir:

This letter submits Special Report 03-01 to notify the NRC of information pertinent to the inservice inspection of steam generator tubes during Crystal River Unit 3 (CR3) Refueling Outage 13 (13R). In accordance with Improved Technical Specification (ITS) 5.7.2.c, Progress Energy, Florida (PEF) is providing, prior to ascension into MODE 4, the following information:

1. Number of tubes plugged and repaired;
2. Crack-like indications and assessment of growth for indications in the first span;
3. Results of in-situ pressure testing; if performed; and
4. Number of tubes and axially oriented TEC (tube end crack-like) indications left in-service, the projected accident leakage, and an assessment of growth for TEC indications.

Special Report 03-01 also provides information to fulfill commitments made in the referenced letter to provide the best estimate leakage that would result from an analysis of the limiting Large Break Loss-of Coolant Accident (LBLOCA) based on as-found circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet re-roll repairs and the ones adjacent to the seal welds.

During the Once Through Steam Generator (OTSG) tube inspection, a Condition Monitoring (CM) Assessment relative to structural and leakage integrity was performed. The CM evaluation concludes that structural requirements were met. As-found values for Main Steam Line Break calculated leakage exceeded the values projected by the previous inspection (Refueling Outage 12) operational assessment. Repairs performed during the 13R OTSG inspection reduced the as-found Main Steam Line Break calculated leakage values to well within the acceptable criteria.

A047

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

This letter establishes no new regulatory commitments.

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

Sincerely,

Sherry L Bernhoft

James H. Terry
Engineering Manager

JHT/lvc

Attachment: Special Report 03-01: Once Through Steam Generator (OTSG) Notifications Required Prior to MODE 4

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

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT

Special Report 03-01
Once Through Steam Generator Notifications Required Prior to MODE 4

ITS 5.7.2.c REPORT

Inservice inspections of the Crystal River Unit 3 (CR3) Once Through Steam Generators (OTSGs) were performed during Refueling Outage 13 (13R). In accordance with Improved Technical Specifications (ITS) Section 5.7, Reporting Requirements,) Progress Energy, Florida (PEF) is submitting Special Report 03-01, for notifications required prior to ascension into MODE 4.

ITS 5.7.2.c states,

> *Following each inservice inspection of the steam generator (OTSG) tubes, the NRC shall be notified of the following prior to ascension into MODE 4:*
>
> 1. *Number of tubes plugged and repaired;*
> 2. *Crack-like indications and assessment of growth for indications in the first span;*
> 3. *Results of in-situ pressure testing, if performed; and*
> 4. *Number of tubes and axially oriented TEC [tube end crack-like] indications left in-service, the projected accident leakage, and an assessment of growth for TEC indications.*

The required information follows:

1. *Number of tubes plugged and repaired (ITS 5.7.2.c.1)*

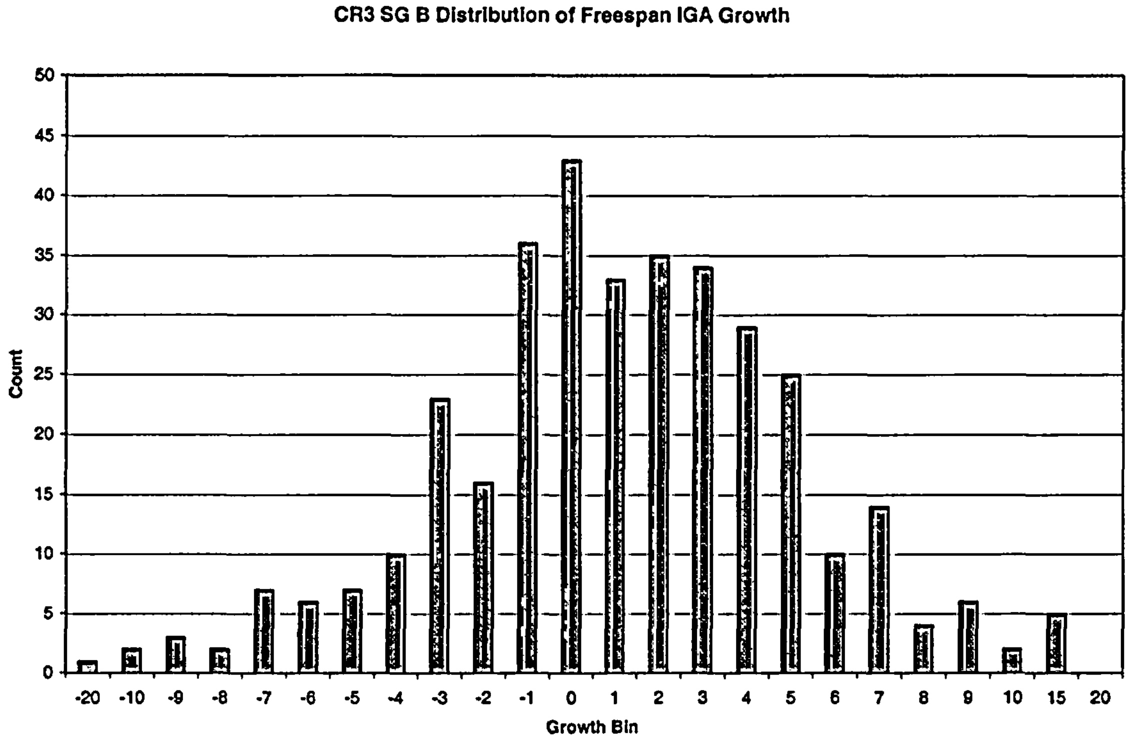
OTSG	NUMBER OF TUBES			
	PLUGGED	RE-ROLLED (UTE)	RE-ROLLED (LTE)	SLEEVED
A	82	177	127	0
B	53	252	13	0

2. *Crack-like indications and assessment of growth for indications in the first span of the "B" steam generator. (ITS 5.7.2.c.2)*

The "first span" is defined as those lengths of tubing which span the region from the secondary face of the lower tubesheet to the first support plate. As required by technical specification 5.6.2.10.4.c, inservice tubes with pit-like intergranular attack (IGA) have been monitored for growth by using a test probe equivalent to the high frequency bobbin probe used in 1997.

The number of crack-like indications in the first span identified during this examination follows:

5 tubes were plugged for >10 % growth.
2 tubes were plugged for ≥40 % Through Wall Dimension.
1 tube was plugged for not meeting the Regression Technique (New Indication)
0 tubes had crack-like indications in the first span.



Based on the distribution of freespan IGA, there is no significant change in the growth rate.

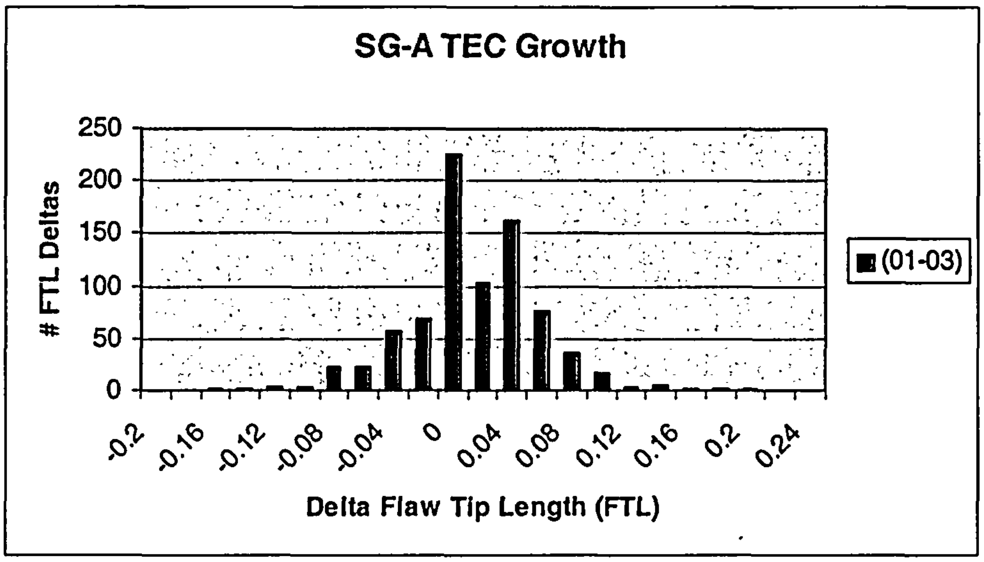
3. *Results of in-situ pressure testing (ITS 5.7.2.c.3)*

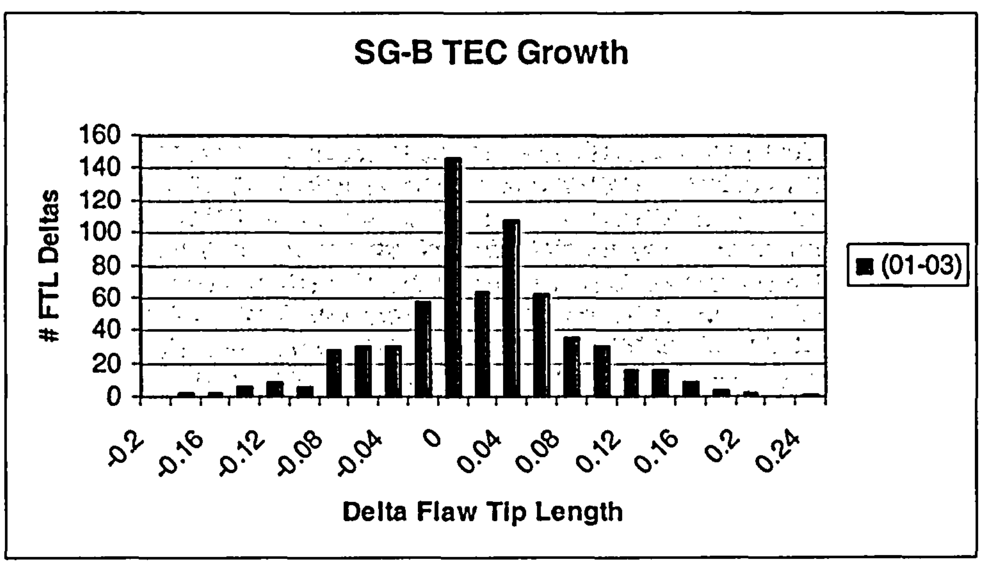
There were no eddy-current indications which exceeded the threshold for performing In-Situ pressure testing. Therefore, no In-Situ pressure test was performed.

4. *Number of tubes and axially oriented TEC indications left in-service, the projected accident leakage, and an assessment of growth for TEC indications (ITS 5.7.2.c.4).*

OTSG	**Tubes**	**Indications**	**Projected Accident Leakage**
Location	UTE / LTE	UTE / LTE	
A	953 / 4	1221 / 7	0.459 gpm
B	729 / 105	959 / 112	0.619 gpm

Note: Projected Accident Leakage equals the Operational Assessment TEC Leakage. This includes the TECs left in service plus the undetected (POD) leakage.





TEC growth rates for both the "A" and "B" OTSGs show there is no significant change in the Flaw Tip Length from 2001 to 2003.

There were no TEC indications that extended past the cladding and into the carbon steel of the tubesheet.

Regulatory Commitments

Following each inservice inspection of steam generator tubes but prior to returning the CR3 steam generators to service, Progress Energy Florida Inc. will verbally notify the NRC of the following:

a. *Number of tubes with circumferential cracking indications inboard of the roll repair.*

   No circumferential indications were identified inboard of the repair rolls.

b. *Number of tubes with circumferential cracking indications in the original roll region, including the zone adjacent to the tube-to-tubesheet seal weld if no re-roll is present.*

   "A" OTSG had 186 circumferential indications (5 in the upper tube ends, 1 in the original roll transition and the remainder were in the lower tube ends.)

   "B" OTSG had 67 circumferential indications (11 in the lower tube ends and 56 in the upper tube ends.)

c. *Determination of the best estimate total leakage that would result from an analysis of the limiting Large Break Loss-of-Cooling Accident (LBLOCA) based on as-found circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet re-roll repairs, and the zones adjacent to the welds.*

   A leakage analysis was performed based on LBLOCA axial tube loading and as-found circumferential cracking indications. The best estimate leak rate from degradation is 7.41 gpm in "A" OTSG, and 3.00 gpm in "B" OTSG.

d. *Demonstrate that the primary-to-secondary leakage following a LBLOCA, as described in Appendix A to Topical Report BAW-2374, Revision 1 is acceptable based on the as-found condition of the steam generators. This is required to demonstrate that adequate margin and defense-in-depth are maintained. For the purpose of this evaluation, "acceptable" means a best estimate of the leakage expected due to a LBLOCA where that leakage would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR Part 50.67 and Part 100 limits).*

The estimated leak rate due to a LBLOCA will not result in a significant increase in radionuclide releases from the plant. This is based on a review of the Steam Generator Tube Rupture (SGTR) assumptions from Chapter 14. The best estimate leakage is less than the Chapter 14 assumptions.

Additional Information

During a telephone conference with the NRC staff on October 28, 2003, the following information was requested:

1. Provide details of TEC leakage and the technical basis prior to Mode 4.

OTSG A TEC Leakage Information				OTSG B TEC Leakage Information			
	Lower	Upper	Total		Lower	Upper	Total
As Found (CM)	0.013	0.932	0.945	As Found (CM)	0.124	1.102	1.226
Repaired	0.000	0.666	0.666	Repaired	0.017	0.824	0.841
Left In Service	0.013	0.266	0.279	Left In Service	0.107	0.278	0.385
Undetected	0.002	0.178	0.180	Undetected	0.024	0.210	0.234
Operational Assessment	0.015	0.444	0.459	Operational Assessment	0.131	0.488	0.619

Including other degradation mechanisms, the total condition monitoring leakage from all sources is 1.029 gpm in S/G A and 1.241 in S/G B. These values are above the 0.856 gpm limit.

The technical basis for acceptability of the method for disposition of TEC leakage resides in Babcock & Wilcox (B&W) Owners Group Topical Report BAW-2346P, Revision 0.