Progress Energy

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

Ref: 10 CFR 50.36

December 3, 2005
3F1205-03

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

Subject: Crystal River Unit 3 - Special Report 05-01: 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:

Pursuant to Improved Technical Specification (ITS) 5.7.2.c, Florida Power Corporation, doing business as Progress Energy Florida, Inc. (PEF) is hereby providing, prior to ascension into MODE 4, Special Report 05-01 to notify the NRC of information pertinent to the inservice inspection of steam generator tubes during Crystal River Unit 3 (CR3) Refueling Outage 14 (14R). The report contains the following:

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.

Additionally, Special Report 05-01 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 zones adjacent to the seal welds.

During the OTSG tube inspection, a Condition Monitoring (CM) Assessment relative to structural and leakage integrity was performed. The CM evaluation concludes that structural and leakage requirements were met. As-found values for Main Steam Line Break calculated leakage were below the values projected by the previous inspection (Refueling Outage 13) operational assessment.

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. Paul Infanger, Supervisor, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

Michael J. Annacone
Engineering Manager

MJA

Attachment: Special Report 05-01: Once-Through Steam Generator 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 05-01
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 14 (14R). In accordance with Improved Technical Specifications (ITS) Section 5.7, Reporting Requirements, Florida Power Corporation, doing business as Progress Energy Florida, Inc. (PEF) is submitting Special Report 05-01, 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] 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 NEW REPAIRS			
	PLUGGED	RE-ROLLED UPPER TUBE END (UTE)	RE-ROLLED LOWER TUBE END (LTE)	SLEEVED
A	52	32	0	0
B	79	64	0	0

In addition to the above new tube repairs, several pre-existing plugs were replaced during the 14R outage as a result of plug maintenance.

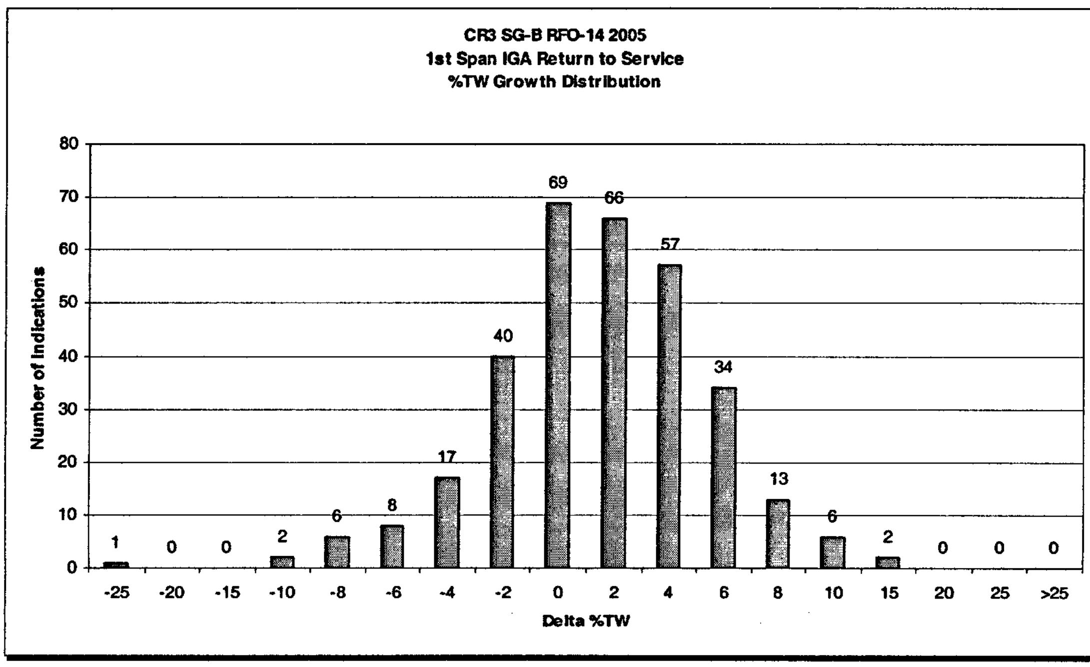
OTSG	PLUG REPLACEMENTS			
	NUMBER OF PLUGS	ORIGINAL PLUG	REPLACEMENT PLUG	COMMENTS
A	1	I-600 Rolled	I-690 Rolled	Roll transition single axial indication (SAI)
B	1	I-600 Welded	I-690 Rolled	Known weld defect
B	2	I-600 Rolled	I-690 Rolled	Roll transition SAI
B	1	I-600 Rolled	I-690 Rolled	Add stabilizer at UTE
B	1	I-600 Rolled	I-690 Welded	Roll transition SAI

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 indications in the first span identified during this examination follows:

0 tubes had crack-like indications for the inservice tubes with pit-like IGA.
0 tubes were plugged for ≥40 % Through Wall Dimension (TWD).
2 tubes were plugged for >10 % growth and ≥ 30% TWD.



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)*

The eddy current inspection identified two (2) indications that did not pass the initial In-Situ pressure test screening criteria based on flaw size and depth. In-Situ test pressures and axial loads were selected based on CR3 design parameters for normal operating differential pressure (NODP) and for accident induced differential pressure and axial loading as required. No leakage was identified during the in-situ pressure tests.

The following table contains information regarding the tubes that were subjected to a full-length pressure test:

OTSG	ROW	TUBE	FLAW TYPE	TEST PRESSURE (psig)	DEPTH (%TW)	LENGTH (inches)	ARC (inches)
A	3	30	Axial	4300	39	4.1"	NA
B	73	55	Circumferential	*5550	70	NA	**0.37"

*includes the pressure for MSLB axial loading.
**conservatively combined the ARC length of two close circumferential indications.

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 UTE / LTE	Indications UTE / LTE	Projected Accident Leakage
A	1282/5	1698/8	0.347 gpm
B	*806/148	1066/161	0.296 gpm

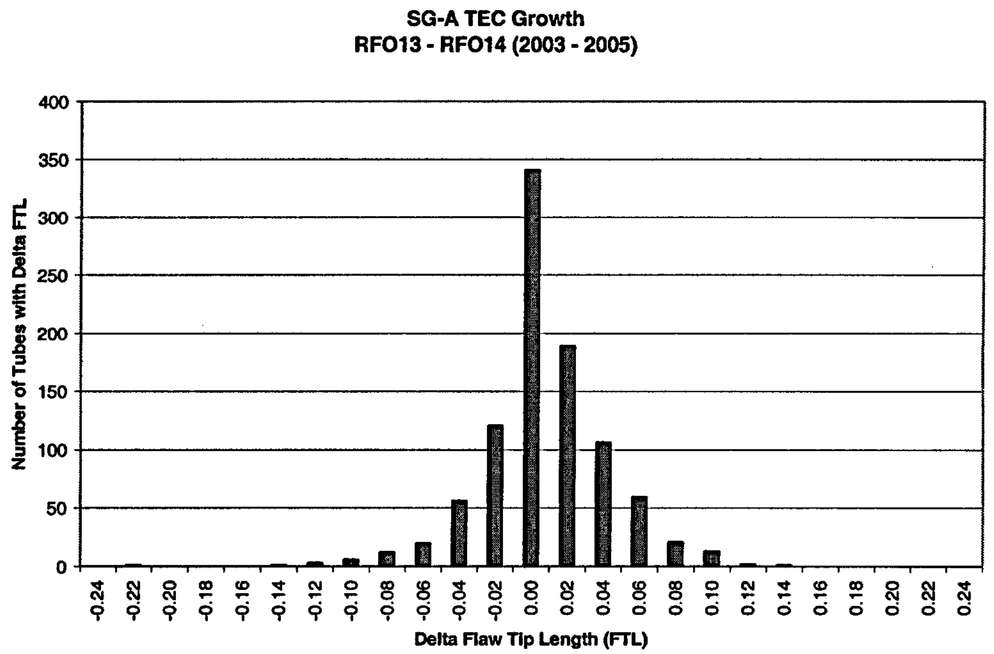
* There were 52 tubes that were duplicates between the upper and lower TEC tubes, resulting in 902 distinct tubes left in service.

Note: Projected Accident Leakage equals the Operational Assessment TEC Leakage. This includes the predicted TEC leakage based on the prediction method described in CR3 License Amendment No. 222.

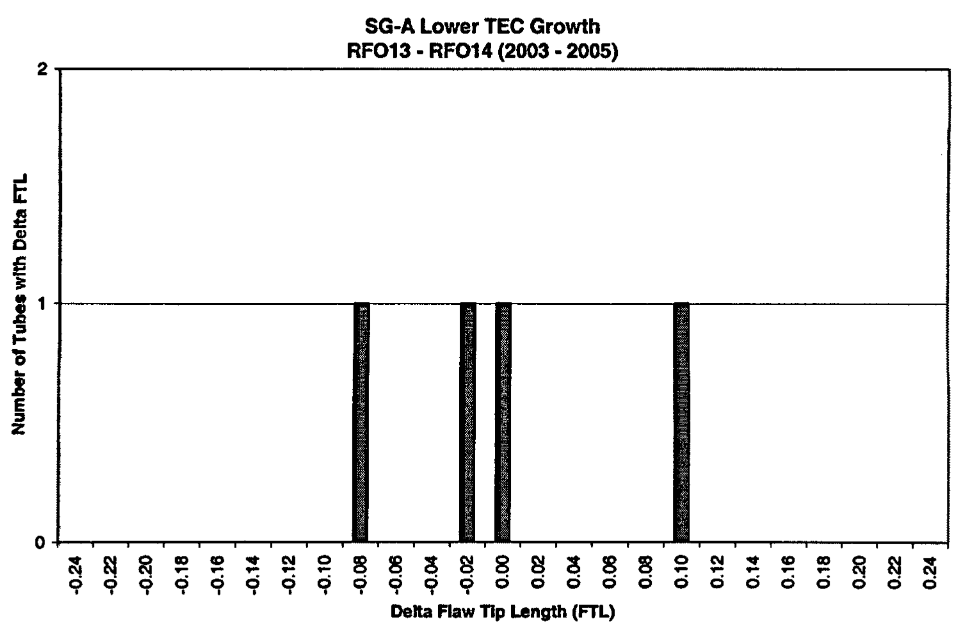
The distance of the flaw tip location (FTL) relative to the carbon-clad interface (CCI) is recorded for each TEC indication. This distance is used to determine the TEC growth rates and to ensure that the indications have not grown into the CCI. For the TEC indications left in service during the previous outage, none had grown into the CCI as confirmed by comparing the previous 13R (2003) and the current 14R (2005) results. Furthermore, none of the new TEC indications had penetrated the CCI.

A distribution of growth rates for the 2003 – 2005 operating cycle is presented below, which provide histograms of the 2003 FTL values minus the 2005 FTL values. Thus, a positive number represents an apparent growth toward the CCI. Based on these figures, the growth distribution may indicate a small amount of positive growth, but nothing of significance. The average growth rate for the upper TEC indications in both steam generators measures -0.002 inch. The average growth rate for the lower TEC indications in both steam generators measures 0.001 inch.

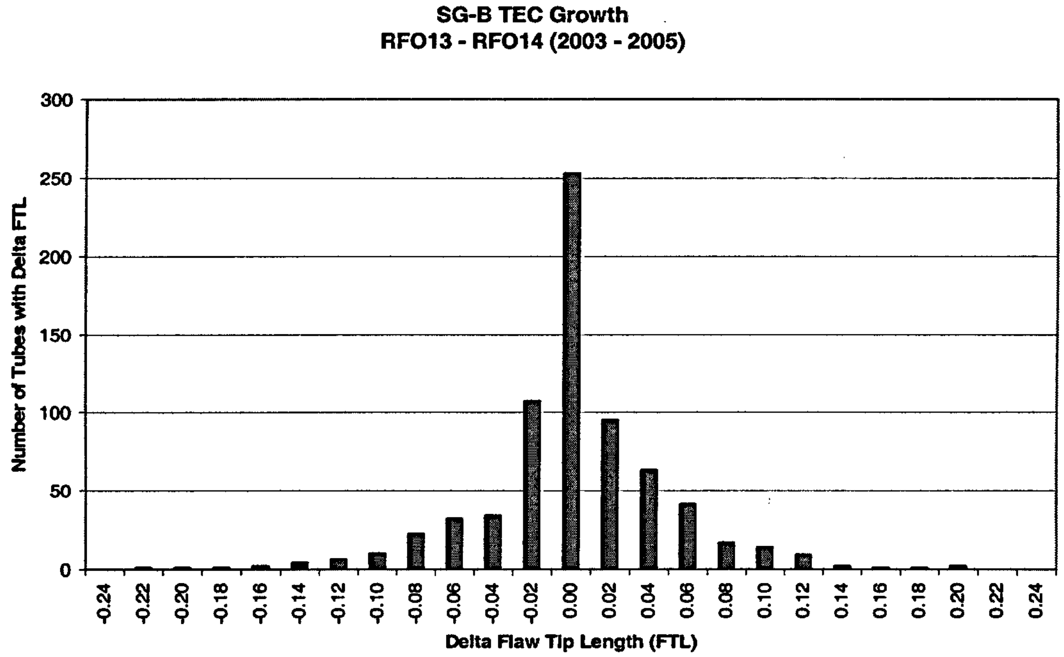
A-OTSG Upper TEC FTL Growth Rate



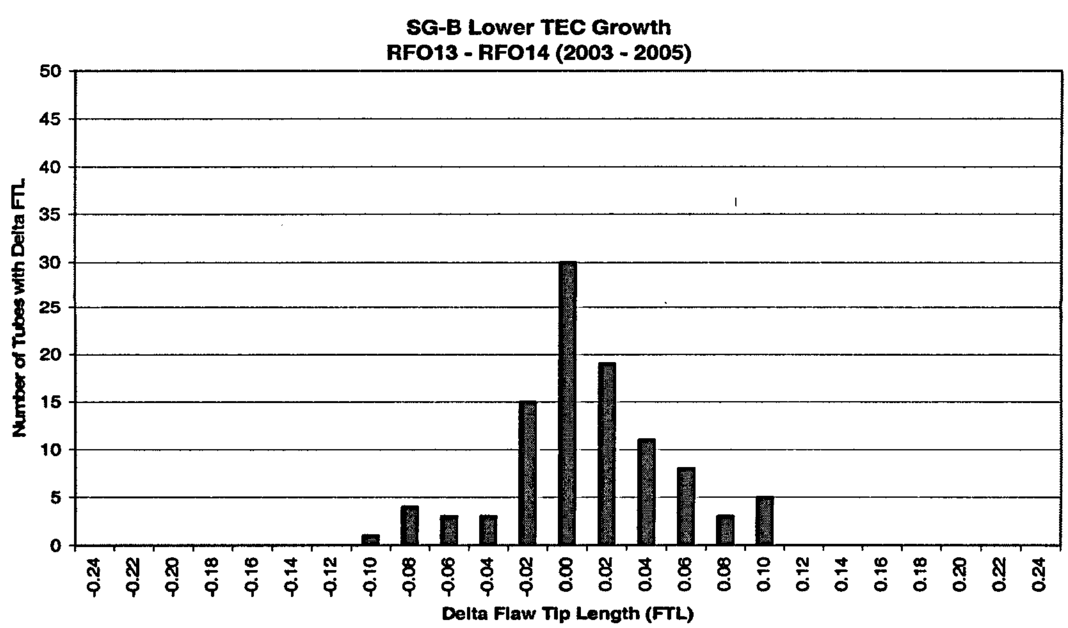
A-OTSG Lower TEC FTL Growth Rate



B-OTSG Upper TEC FTL Growth Rate



B-OTSG Lower TEC FTL Growth Rate



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.*

In the A OTSG, 0 tubes had circumferential cracking inboard of the re-roll.
In the B OTSG, 1 tube had circumferential cracking inboard of the re-roll.

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 34 tubes with circumferential cracking indications (3 tubes in the upper tube sheet [UTS] and 31 tubes in the lower tube sheet [LTS])

B-OTSG had 45 tubes with circumferential cracking indications (40 tubes in the UTS and 5 tubes in the LTS)

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 5.5 gpm in "A" OTSG, and 3.3 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 would not have resulted in a significant increase in radionuclide releases from the plant. This is based on a review of the Steam Generator Tube Rupture (SGTR) accident assumptions from Chapter 14 (Safety Analysis) of the FSAR. The LBLOCA best estimate leakage is less than the FSAR assumptions for SGTR which assumed a flow rate to the secondary side of the affected steam generator to be 435 gpm. The conservative SGTR accident leak rate and assumptions for degraded fuel cladding only result in a 2 hour integrated dose at the exclusion area boundary of 0.139 rem compared to a limit of 2.5 rem. Therefore the LBLOCA leakage of 5.5 gpm for the "A" OTSG and 3.3 gpm for the "B" OTSG is bounded by the FSAR evaluation.