November 23, 2005

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing and
Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: REVIEW OF CRYSTAL RIVER, UNIT 3, STEAM GENERATOR TUBE INSERVICE INSPECTION SUMMARY REPORTS FROM THE FALL 2003 OUTAGE (TAC NOS. MC1176 AND MC1853)

Dear Mr. Young:

By letters dated October 31, 2003 (ADAMS Accession No. ML033090110), January 27, 2004 (ML040350037), August 10, 2004 (ML042320561), September 9, 2004 (ML042710359), October 27, 2004 (ML043060425), November 22, 2004 (ML043340228), November 24, 2004 (ML043350045), March 30, 2005 (ML051020360), May 20, 2005 (ML051520535), and July 8, 2005 (ML051940269), Florida Power Corporation (the licensee, also doing business as Progress Energy Florida, Inc.) submitted information pertaining to the steam generator tube inspections at Crystal River Unit 3 during its fall 2003 refueling outage (designated 13R). Additional information concerning these inspections was summarized by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter dated February 17, 2004 (ML040490002). In addition, a public meeting was held on April 26, 2005, to discuss, in part, 2003 steam generator tube inspection results (ML051190330).

As discussed in the enclosed evaluation, the NRC staff concludes that the licensee provided the information required by its technical specifications. In addition, the NRC staff did not identify any technical issues that warrant followup action at this time; however, the NRC is still evaluating (through the reactor oversight process) the corrective actions taken by the licensee in response to determining that it did not meet the accident induced leakage limit.

This completes the NRC staff's efforts under TAC Nos. MC1176 AND MC1853. If you have any questions regarding this matter, please contact me at (301) 415-2020.

Sincerely,

/RA/
Brenda L. Mozafari, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosure: As stated

cc w/enclosure: See next page

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EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE STEAM GENERATOR TUBE INSPECTION REPORTS FOR THE 2003 OUTAGE

RELATED TO THE FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3 NUCLEAR POWER PLANT

DOCKET NO. 50-302

By letters dated October 31, 2003 (ADAMS Accession No. ML033090110), January 27, 2004 (ML040350037), August 10, 2004 (ML042320561), September 9, 2004 (ML042710359), October 27, 2004 (ML043060425), November 22, 2004 (ML043340228), November 24, 2004 (ML043350045), March 30, 2005 (ML051020360), May 20, 2005 (ML051520535), and July 8, 2005 (ML051940269), Florida Power Corporation (the licensee, also doing business as Progress Energy Florida, Inc.) submitted information pertaining to the steam generator tube inspections at Crystal River Unit 3 (CR-3) during its fall 2003 refueling outage (designated 13R). Additional information concerning these inspections was summarized by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter dated February 17, 2004 (ML040490002). In addition, a public meeting was held on April 26, 2005, to discuss, in part, 2003 steam generator tube inspection results (ML051190330).

CR-3 has two Babcock and Wilcox once-through steam generators. Each steam generator contains 15,531 stress relieved, mill annealed, Alloy 600 tubes. Each tube has a nominal outside diameter of 0.625 inch and a nominal wall thickness of 0.034 inch. The tubes were mechanically roll expanded in both the hot- and cold-leg tubesheet for approximately 1 inch of the 24-inch thick tubesheets. The tubes are supported by a number of carbon steel support plates. The hot-leg temperature is approximately 603 degrees Fahrenheit. The steam generators had operated for 17.6 effective full-power years as of October 2003.

The NRC has approved a number of amendments related to the CR-3 steam generators. The licensee implements a tube end cracking (TEC) alternate tube repair criteria, is permitted to repair tubes by re-rolling them in the tubesheets, is permitted to repair tubes by sleeving, and depth sizes intergranular attack indications in the first span of steam generator B. Approximately 160 Alloy 690 sleeves are installed in the lane/wedge region of each steam generator. All sleeves were installed in 1994. Approximately 935 tubes in steam generator A and 1355 tubes in steam generator B have re-rolls installed in the tubesheet region. There are re-rolls in both the upper and lower tubesheet. There are approximately 6000 tubes in the kidney region (a region above the lower tubesheet where the sludge height is greater than or equal to 1 inch) of steam generator A and approximately 4100 tubes in the kidney region of steam generator B. The licensee estimates the size of intergranular attack indications along the length of the tube between the secondary face of the lower tubesheet and the first tube support plate (i.e., in the first span) in steam generator B.

The licensee provided the scope, extent, methods, and results of its steam generator tube inspections in the documents referenced above. The licensee also described corrective actions (i.e., tube plugging or repair) taken in response to the inspection findings. In addition, the licensee reported that based on the 13R (2003) inspection results, they determined the analyzed limit on the amount of primary-to-secondary leakage assumed to occur during postulated accident conditions had been exceeded. As a result of the review of the previously referenced reports, the NRC staff developed the following comments/observations:

1. Cracks have been observed to initiate in wear scars at several plants. In addition, the ability to distinguish a crack from a wear scar based on bobbin coil data is limited. Since bobbin indications attributed to wear scars are sized and left in service, if the bobbin indication is actually a result of a crack within a wear scar (or is simply a crack), the size of the crack (or crack within a wear scar) may be underestimated since the method for sizing a crack and wear may differ. If the flaw is severe enough, it may no longer meet the structural integrity performance criteria at the end of the next operating interval. Therefore, some plants inspect all (or a significant sample) of bobbin indications which they suspect are a result of wear (or had confirmed as wear during a previous outage) with a rotating probe each outage (even if the bobbin coil data shows no change). These rotating probe examinations are performed to confirm that the bobbin indication is actually a result of wear and not some other degradation mechanism (e.g., cracks or a combination of a crack within a wear scar).

   Of the suspected wear indications detected at CR-3 in 2003 with a bobbin coil, a sample of tubes was inspected with a rotating probe. This sample included all newly identified potential wear indications (i.e., those bobbin indications expected to be a result of wear for which no previous rotating probe examinations were performed to identify the nature of the degradation) and any bobbin signal (previously attributed to wear based on a rotating probe examination) that had a notable change. As a result of these inspections, no cracking in a wear scar has been observed. Although no crack-like indications have been detected in wear scars at CR-3, the operating experiences at other plants indicate the potential for cracks to develop in wear scars. Given the goal to ensure tube integrity for all tubes, it is important to have a high degree of confidence that indications are appropriately classified (e.g., as either wear, cracking, or cracking in combination with wear) so that they are appropriately sized and dispositioned. In addition, it is important for tube integrity analyses to reflect the potential for this degradation mechanism to occur.

2. The licensee's response to Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections" is being reviewed separately. As a result, this review does not address the specific issues raised in the generic letter.

3. As mentioned above, the licensee determined that its accident induced primary-to-secondary leakage limit of 0.856 gallons per minute (gpm) was exceeded during the 13R outage. The fundamental reason for exceeding the leakage limit was that the licensee under predicted the number of TEC indications that they would detect. In response to requests for additional information, the NRC staff was not able to identify any analysis performed by the licensee prior to restart to confirm the adequacy of the

plugging and repairs performed during 13R to ensure that the accident induced leakage limit would not be exceeded in 14R. The NRC notes that one revised method intended to address the NRC's questions was subsequently provided in the licensee's March 30, 2005, letter; however, this method was subsequently withdrawn by the licensee's May 20, 2005, letter. In addition, another analysis was provided as part of an amendment request (in response to an NRC request for additional information) by letter dated August 12, 2005 (ML052410111). This methodology was reviewed by the NRC staff as part of the license amendment process. A revised amendment dated September 9, 2005, provided a revised methodology that was approved by Amendment No. 222, dated October 31, 2005.

Given that the licensee did not provide an analysis to confirm the adequacy of the repairs performed during 13R, the NRC performed its own assessment using the data provided in the licensee's May 20, 2005 letter. In performing its assessment, the NRC staff determined that the methodology used for projecting the number of new TEC indications has consistently underpredicted the number of indications since implementation of this alternate repair criteria (actually since the 2001 (12R) inspection, which was the first inspection in which a comparison between the projected and actual number of indications could be performed).

Based on the earlier assessment, the NRC staff could not conclude that the accident induced leakage limit would be met during 14R. Using various methods, the NRC determined that the accident induced leakage from TEC indications for the most limiting steam generator could be as high as 1.2 gpm with a best-estimate slightly above the leakage limit of 0.856 gpm. In addition, the NRC staff also projected the amount of leakage using the methodology that the licensee proposed in its August 12, 2005, submittal (and the leakage values per indication approved for use during the 2003 outage). This methodology also projects the accident induced leakage limit will be exceeded during 14R. The revised methodology was accepted in Amendment No. 222. An assessment of the licensee's corrective actions in response to exceeding the accident induced leakage limit is being conducted under NRC inspection 50-302/2005009.

4. In 1999, approximately 35 circumferential indications were found at dents near the secondary face of the lower tubesheet. Since 1999, no additional circumferential indications were reported near the lower tubesheet. Operating experiences at other plants generally shows that once a corrosion mechanism initiates, additional indications are generally found in subsequent inspections. In addition, operating experience has shown that denting can limit the ability to detect degradation. As a result, by not routinely inspecting all dented/dinged locations with a probe capable of finding the forms of degradation potentially affecting the tube at these locations, tube integrity analysis may become a challenge. This consideration has led many licensees to inspect all dented/dinged locations greater than some voltage threshold with a rotating probe to confirm the absence of cracking at these locations.

Based on a review of the information provided (as discussed above), the NRC staff concludes that the licensee provided the information required by its technical specifications. In addition, with the exception of the NRC's assessment of the corrective actions taken by the licensee in response to determining that they did not meet the accident induced leakage limit (which is being conducted under the Reactor Oversight Process), the NRC staff concludes that there are no technical issues that warrant followup action at this time.

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