June 19, 2008

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

SUBJECT: CRYSTAL RIVER UNIT 3 – SUMMARY OF CONFERENCE CALLS REGARDING THE 2007 STEAM GENERATOR TUBE INSPECTIONS (TAC NO. MD7276)

Dear Mr. Young:

On November 16 and 26, 2007, the Nuclear Regulatory Commission (NRC) staff participated in conference calls with representatives of Florida Power Corporation (licensee) regarding steam generator tube inspections performed at Crystal River, Unit 3 (CR-3), during the fall 2007 refueling outage. To facilitate this conference call, the licensee provided written materials regarding CR-3's refueling outage 15 steam generator eddy current inspections and results (Agencywide Documents Access and Management System Accession Nos. ML080150467 and ML01504960). Enclosed is a summary of the conference calls.

Based on the information provided during the conference calls, the NRC staff did not identify any issues that warranted additional follow-up action at this time.

This completes the staff's review of the preliminary results for the 2007 steam generator tube inspections at CR-3. If you have any questions regarding this matter, please contact me at (301) 415-1447.

Sincerely,

/RA/

Farideh E. Saba, 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/encl: See next page

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SUMMARY OF NOVEMBER 16 AND 26, 2007, CONFERENCE CALLS

STEAM GENERATOR TUBE INSPECTIONS

CRYSTAL RIVER, UNIT 3

DOCKET NO. 50-302

On November 16 and 26, 2007, the Nuclear Regulatory Commission (NRC) staff participated in conference calls with representatives of Florida Power Corporation (licensee), regarding steam generator tube inspections performed at Crystal River, Unit 3 (CR-3), during the fall 2007 refueling outage. CR-3 is on a 24-month operating cycle. This was the last refueling outage and steam generator (SG) inspection before the SGs are replaced in 2009. At the time of the call on November 16, the eddy current inspections were approximately 80 percent complete.

CR-3 has two Babcock & Wilcox once-through steam generators (OTSG), designated OTSG-A and OTSG-B. Each SG contains 15,513 stress relieved, mill annealed Alloy 600 tubes. Each tube has a nominal diameter of 0.625 inch and a nominal thickness of 0.034 inch. The tubes were mechanically roll expanded in the tubesheets. The tubes are supported by a number of carbon steel supports.

Prior to the November 16 call, the licensee provided written materials regarding CR-3's refueling outage 15 SG eddy current inspections and results, which can be accessed from the NRC's Agencywide Documents Access and Management System (ADAMS) through accession numbers ML080150467 (Enclosure 2) and ML01504960 (Enclosure 3). In Enclosure 2, the licensee provided information to fulfill three commitments that were identified in the NRC letter to the licensee, "Crystal River Unit 3 – Issuance of Amendment Regarding Reroll Repair for Once-Through Steam Generator," dated September 10, 2001. Enclosure 3 contains the licensee's responses to the questions provided in NRC's November 8, 2007 email, "talking points for SG conference call." At the time of the call on November 16, the eddy current inspections were approximately 80 percent complete. No unusual degradation or unexpected conditions had been detected at the time of the November 8, 2008 call. A summary of the discussions during this call is provided below:

- Based on the trends provided on Page 3 of Enclosure 3, the primary to secondary leakage remained steady during the operating cycle 15. It is noted that the leakage rate is shown as gallons per day. The three spikes shown on the plot provided by the licensee were due to reactor power level changes and are not indications of actual leakage changes.

- There are some tubes with restrictions near the tube end. The restrictions were caused by a loose part in a prior cycle. The licensee does not expect to have to inspect these restricted tubes with a smaller diameter bobbin probe since they should be able to inspect the tubes from the other tube end.

- The lower tubesheet motorized rotating pancake coil (MRPC) inspections were performed at least 4 inches above the tubesheet to be sure they encompassed the sludge pile height.

- During the last inspection, in 2005, the licensee inspected 100 percent of the tubes in both OTSG-A and B tubesheets with an MRPC. No indications were identified in the lower tubesheet region in OTSG-A, but there were some indications in the OTSG-B lower tubesheet. Therefore, during this outage only 20 percent of the tubes in the lower tubesheet of OTSG-A were inspected while 100 percent of the tubes in OTSG-B were inspected with an MRPC.

- On page 11 of Enclosure 3, the date should be 11/15/07.

- The licensee identified approximately 2300 multiple axial anomalies (MAA) in the OTSGs. This is similar to what was observed in the last outage. The number of MAA indications reported is not the number of tubes with this type of indication. There could be multiple indications in a single tube. These indications are in the tube end region.

- The axial indication in the freespan in OTSG-A was attributed to intergranular attack (IGA) associated with grooves. During the last outage, several tubes with the worst indications were pressure tested. No leakage or burst occurred during these tests. The circumferential indications described on Page 14 of Enclosure 3 are all due to primary water stress corrosion cracking. Two indications were identified as outside diameter stress corrosion cracking possibly because these indications had a phase angle that was representative of an outside diameter indication.

- No indications had been detected in tube sleeves in the lower tubesheet crevice region, or in the lane/wedge region at the time of the call.

- There were two volumetric/wear indications reported. The through wall depth was reported as 59 percent. This indication is due to IGA, not wear.

- Unacceptable indications in the upper tubesheet transition rolls will be repaired by rerolling. Indications that cannot be re-rolled will be plugged as required.

At the time of the November 16 call, there was no in-situ pressure tests planned. However, during the call on November 26, the licensee reported that there were three in-situ tests performed on freespan axial indications. There was no leakage observed during the in-situ tests and no tubes burst.

A foreign object was identified inside tube 73-99 in OTSG-A, 0.5 inch down from the upper tube end. It had not yet been retrieved at the time of the November 16 call. If it can not be retrieved, then the affected tube will be plugged.

Another issue involving loose items on the primary side was discussed by the licensee. One of the eddy current probes failed when a spot weld broke. This allowed three small pins (0.1 inch long by 1/16 inch diameter) to back out of the probe body. These pins hold the coils in place. At the time of the call, one pin had not yet been recovered. If the pin is not found, then an evaluation will be performed to determine the acceptability of leaving this pin in the reactor coolant system. A new rotating probe design from the vendor, which eliminated the spot weld, was going to be used to complete the inspections. The susceptibility of these probes to fail had been previously reported by Three Mile Island.

A follow up conference call was held on November 26, 2007. The purpose of the call was to fulfill a CR-3 licensing commitment to notify the NRC staff of certain indications in the tubesheet related to roll repairs, original roll region, and total leakage estimates due to indications in the tubesheet (for more details see ADAMS Accession No. ML012270329). The following is a summary of the information provided by the licensee during this call:

- The quantity of tubes with circumferential cracking inboard of roll repairs is consistent with the last inspection in 2005.

- Most of the circumferential cracking indications in the original roll region are near the tube ends (i.e., within 0.5 inch of the tube end), not in the rolled region. The number of these indications is higher than what was observed in the last outage. The circumferential crack indications are repaired by rerolling if possible/practical.

- The leakage estimates during a loss-of-coolant accident were slightly higher than what was determined during the last outage.

- No circumferential crack indications were found in the freespan. The leakage estimates includes leakage from cracks outside the reactor coolant pressure boundary.

The NRC staff did not identify any issues or safety concerns that required follow-up action during the November 26, 2007, conference call; however, the staff asked to be notified in the event that any unusual conditions were detected during the remainder of the outage.