September 7, 2005

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION REGARDING THE 15-DAY AND 90-DAY STEAM GENERATOR TUBE INSERVICE INSPECTION REPORTS FOR THE END-OF-CYCLE 6 REFUELING OUTAGE IN 2005 (TAC NO. MC7485)

Dear Mr. Singer:

By letter dated March 23, 2005, Tennessee Valley Authority (TVA) submitted the 15-day steam generator (SG) plugging and sleeving report for the spring 2005 outage at Watts Bar Unit 1 in accordance with Technical Specification (TS) 5.9.9. By letter dated March 23, 2005, TVA submitted the F* alternate repair criteria report in accordance with TS 5.9.9. By letter dated June 28, 2005, TVA submitted the 90-day SG voltage-based alternate repair criteria report. In addition to these reports, the U.S. Nuclear Regulatory Commission staff summarized additional information concerning the 2005 SG tube inspection in a letter dated June 20, 2005.

In order for the staff to complete its review of the above reports, we request that you provide responses to the enclosed request for additional information.

Sincerely,

/RA/
Douglas V. Pickett, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosure: Request for Additional Information

cc w/enclosure: See next page

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Mr. Karl W. Singer
Tennessee Valley Authority

WATTS BAR NUCLEAR PLANT

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REQUEST FOR ADDITIONAL INFORMATION

WATTS BAR END-OF-CYCLE 6 STEAM GENERATOR

INSERVICE INSPECTION REPORTS

TAC NO. MC7485

1. Subsequent to your 2003 steam generator tube inspections (End-of-Cycle 5), you identified five tubes which were left in service despite having indications in the parent tube below the location where a sleeve was installed during the 2003 outage. Please confirm that these five tubes (row 18 column 35; row 19 column 32; row 22 column 31, row 22 column 37, row 42 column 55) were plugged during your 2005 outage (end-of-cycle 6). In addition, please confirm that no similar tubes were left in service following your 2005 steam generator tube inspections (i.e., confirm that F* was not applied to a sleeved tube).

2. In your June 28, 2005 letter, it was indicated that several of the tubes remaining in service contained indications located where the tube passes through the flow distribution baffle. Please confirm that your amendment request for implementing the voltage-based tube repair criteria discussed in Generic Letter (GL) 95-05 addressed the conditions in Section 2.a.3 of Attachment 1 to GL 95-05 (since specific U.S. Nuclear Regulatory Commission (NRC) approval is needed to apply the voltage-based limits to flow distribution baffle intersections). That is, confirm that your amendment request addressed the causal factors for high voltage growth at flow distribution baffle intersections and the applicability of these conditions at your plant. Also, please discuss whether the average growth rates for the flow distribution baffle indications were less than that observed in steam generator 3 (although page 3-18 indicates that the average growth rates are less than that seen in steam generators 1 and 2, flow distribution baffle indications were found in steam generators 1, 2, and 3). If not, please discuss the implications.

3. One tube was identified with a 6.32 volt indication. Please discuss what actions, if any, were taken to ensure that this tube had adequate structural integrity since the voltage exceeded the structural limit of 5.65 volts.

4. On page 5-7 of your June 28, 2005, letter, you indicated that the end-of-cycle 6 predicted values of the probability of burst and leakage were conservative because they were based on a very conservative industry voltage growth rate. However, your January 15, 2004, letter (ML040220171) indicates that Table 3-9 was used for the end-of-cycle 6 projections. Table 3-9 provided a bounding growth rate from all four steam generators at Watts Bar. Please confirm whether an industry voltage growth rate or a plant-specific growth rate was used in the calculations.

5. The largest voltage indication observed at the end-of-cycle 6 (2005) was not predicted. This resulted in exceeding the limiting projection on probability of burst. In addition, the most limiting accident induced leak rate (0.175 gallons per minute (gpm)) was exactly predicted (although for a different steam generator). Given that the maximum voltage indication was not predicted and that your probability of burst projections for

Enclosure

end-of-cycle 7 (8.65×10^{-3}) are near the limits (10^{-2}), discuss what corrective actions were taken to ensure such an under-prediction in the maximum observed voltage (and probability of burst) does not occur for the end-of-cycle 7. The NRC recognizes that tube inspections will not be performed at the end-of-cycle 7 due to the planned replacement of the Watts Bar steam generators.

6. Figure 3-17 indicates that several of the larger voltage indications exhibited negative growth. Although the staff is aware that this does occur at other plants, the large number of negative growths (resulting in a decreasing trend in growth rate as a function of beginning-of-cycle voltage) does not appear to be consistent with that observed at other plants. Please discuss any insights on this trend including a discussion of whether these results draw into question the non-destructive examination uncertainty models applied at Watts Bar.

7. In section 4.6, the upper voltage repair limit was calculated. Please discuss the purpose for the “(518/482)” adjustment in calculating this limit.

8. Section 5.0 describes the condition monitoring assessment. Figures 5-1 through 5-4 depict the distribution of end-of-cycle voltages adjusted by the non-destructive examination uncertainty distribution. Please discuss whether the discrete distributions in these figures (which may have been truncated/adjusted for fractional indications) were used in the condition monitoring assessment or whether the condition monitoring assessment utilized a non-truncated/adjusted distribution of indications.

9. Section 4.7 provides an assessment of the probe wear criteria used at Watts Bar. In a portion of this assessment, the ratio of the current number of indications greater than 1 volt to the total number of these indications that were inspected with a worn probe in the previous inspection was compared to the ratio of the number of indications greater than 1 volt to the total number of indications in the current inspection. Based on similar ratios, a conclusion was drawn that there was no significant effect of probe wear on the population of indications. The staff notes that such a comparison is only valid if the number of tubes inspected with both a worn and good probe is comparable. That is, if the number of tubes inspected with worn probes is significantly different from the number of tubes inspected with a good probe, an erroneous conclusion may be made with respect to the adequacy of the probe wear criteria. As a result, please compare the percentage of new indications at end-of-cycle (EOC) 6 that were inspected with a worn probe during the EOC 5 inspections to the percentage of new indications that were inspected with a good probe during the EOC 5 inspections. In addition, please compare the percentage of new indications greater than or equal to 0.5 volts during the EOC 6 inspections that were inspected with a worn probe during the EOC 5 inspections to the percentage of new indications greater than or equal to 0.5 volts during the EOC 6 inspections that were inspected with a good probe during the EOC 5 inspections. If there are significant differences, please provide an assessment of the adequacy of the probe wear criteria and its impact on your operational assessment for EOC 7. Calculation of the above percentages requires the total number of tubes inspected with a good and worn probe during the EOC 5 inspections. A value of 0.5 volts was chosen to be consistent with the NRC staff’s approval of the alternate probe wear criterion (refer to NRC letter to the Nuclear Energy Institute dated February 9, 1996).