UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 17, 2009

Mr. Preston D. Swafford
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING STEAM GENERATOR TUBE INSERVICE INSPECTION REPORT FOR THE CYCLE 15 REFUELING OUTAGE (TAC NO. MD9595)

Dear Mr. Swafford:

By letters dated August 27, 2008, and November 21, 2008, Tennessee Valley Authority (TVA) submitted information summarizing the results of the Spring 2008 steam generator (SG) tube inspections performed at Sequoyah Nuclear Plant Unit 2 during the Cycle 15 refueling outage.

In order for the staff to complete its review of the information provided, we request that TVA provide responses to the enclosed request for additional information (RAI). Based on discussions with your staff, we understand that you plan to respond to the enclosed RAI by April 15, 2009.

If you have any questions about this material, please contact me at (301) 415-2788.

Sincerely,

Tracy J. Orf, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-328

Enclosure: Request for Additional Information

cc: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

STEAM GENERATOR TUBE INSERVICE INSPECTION REPORT

FOR THE CYCLE 15 REFUELING OUTAGE

SEQUOYAH NUCLEAR PLANT, UNIT 2

By letters dated August 27, 2008, and November 21, 2008, Tennessee Valley Authority (TVA) submitted information summarizing the results of the spring 2008 steam generator (SG) tube inspections performed at Sequoyah Nuclear Plant Unit 2 during the Cycle 15 refueling outage. In order for the staff to complete its review, the following additional information is requested:

1. Please discuss the results of your foreign object search and retrieval (FOSAR). In particular, please discuss whether any loose parts (foreign objects) were left in the steam generator following FOSAR. If so, please confirm that an analysis was performed that demonstrated tube integrity would be maintained until the next steam generator tube inspection. Please discuss whether the locations where possible loose part indications (detected during the eddy current inspection) were inspected visually. If not, please discuss how the tubes near these possible loose parts were dispositioned.

2. Please discuss the scope and results of your secondary side steam drum inspections.

3. On page 6 of 106 in your November 21, 2008 letter, you indicated under the heading "ODSCC [outside diameter stress corrosion cracking] Circumferential at HTS [top of tubesheet hot-leg]" that all axial indications met the condition monitoring performance criteria and all were stabilized and plugged. Please confirm that this was intended to read all circumferential indications met the performance criteria and were stabilized and plugged.

4. Your condition monitoring addresses various degradation mechanisms. However, several degradation mechanisms are not discussed (e.g., axially and circumferentially-oriented outside diameter initiated stress corrosion cracking in the U-bend and at dings). Please clarify why these degradation mechanisms were not listed. (Is it because the previous operational assessment did not predict that these types of indications would be present?)

5. For several degradation mechanisms (e.g., freespan cracking, circumferentially-oriented outside diameter initiated stress corrosion cracking near the top of the hot-leg tubesheet), the number of indications projected to be present was under predicted. Please discuss whether your operational assessment methodology was modified to account for this under prediction in the number of indications.

6. Several primary water stress corrosion cracking indications were found in the U-bend region of steam generator 2. Although the inspection sample was expanded to include all of the affected tubes in steam generator 2, similar expansions were not performed in

the other three steam generators. Please confirm that the operational assessment (until U2C16) for the three steam generators in which 100 percent of the potentially susceptible tubes were not inspected accounted for the potential that a flaw similar (or larger) in size to what was detected in steam generator 2 was present. If the operational assessment did not account for this potential, please discuss why [given that the previous (pre-2008) inspections in this region in all steam generators were comparable].

7. One tube was detected with several indications of stress corrosion cracking near the tube end. This tube was inspected with a rotating probe because it was not fully expanded in the tubesheet region. Please confirm that all of the indications in this tube were detected in the region of the tube where the previously installed plug was expanded into the tube (i.e., the plug expansion zone). In particular, address whether the indications in this tube are attributed to the previous plugging process that this tube was exposed to or whether the stress corrosion cracking is a result of the "normal" fabrication expansion process (i.e., tack expansion). The staff notes that the mechanism responsible for cracking is important in determining which tubes are susceptible to this cracking mechanism (and therefore are required to be inspected).

8. Regarding the one tube plugged because of the "inability to perform a rotating probe examination in the U-bend region," please discuss how it was confirmed that this tube did not have a defect. If the tube could not be inspected with a rotating probe, was an in-situ pressure test performed to confirm that the tube had adequate integrity?

9. Please confirm that you did not use the probability of prior cycle detection model in your assessment of tube integrity for outside diameter stress corrosion cracking at the tube support plate elevations.

10. Tubes with indications greater than 1.5 volts that were inspected with a worn probe were re-inspected. Some of these retested tubes also had indications that were less than 1.5 volts. In these cases, the original voltage (i.e., with the worn probe) was used in your tube integrity analysis. Please discuss why this approach was used rather than using the voltage obtained from a probe that passed the probe wear check. In addition, please provide a listing of the indications including the "worn probe" voltage and the voltage obtained from the probe that passed the probe wear check.

11. On page B-1 of your August 27, 2008 letter, you indicated that "The leak rates were not significantly affected by including the uncertainties in the ANL [Argonne National Laboratory] tearing model." Please clarify this sentence (e.g., was an assessment of the leak rate for this indication performed using the ANL model and were the results with and without uncertainties comparable).

12. In your operational assessment and condition monitoring assessment, 0.008 gallons per minute primary to secondary leakage was attributed to "all other sources." Please clarify the nature of these sources (e.g., cracking in U-bend).

13. On page E2-2 of your August 27, 2008 letter, you indicated that primary water stress corrosion cracking indications at the top of the tubesheet were included in the condition monitoring W* leakage evaluation regardless of whether or not they were above the bottom of the WEXTEX transition (BWT). You then indicate that: "The location of upper

crack tip was subtracted for the location of the BWT and then this value had the non-destructive examination (NDE) uncertainty subtracted. If the value was negative, it was then assumed to be zero." Please clarify these last two sentences.

14. In the third paragraph on page E2-2 of your August 27, 2008 letter, you indicate that "The leakage value for the bins was summed to obtain the total in the 8 inches to 12 inches below the top of tubesheet region." Should this sentence have indicated that this was the leakage for the 0-inch to 8-inch region?

March 17, 2009

Mr. Preston D. Swafford
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING STEAM GENERATOR TUBE INSERVICE INSPECTION REPORT FOR THE CYCLE 15 REFUELING OUTAGE (TAC NO. MD9595)

Dear Mr. Swafford:

By letters dated August 27, 2008, and November 21, 2008, Tennessee Valley Authority (TVA) submitted information summarizing the results of the Spring 2008 steam generator (SG) tube inspections performed at Sequoyah Nuclear Plant Unit 2 during the Cycle 15 refueling outage.

In order for the staff to complete its review of the information provided, we request that TVA provide responses to the enclosed request for additional information (RAI). Based on discussions with your staff, we understand that you plan to respond to the enclosed RAI by April 15, 2009.

If you have any questions about this material, please contact me at (301) 415-2788.

Sincerely,

/RA/

Tracy J. Orf, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-328

Enclosure: Request for Additional Information

cc: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPL2-2 r/f
RidsNrrPMTOrf
RidsNrrPMBMoroney
RidsNrrLACSola
RidsNrrDciCsgb
RidsOgcRp
RidsAcrsAcnw_MailCenter
RidsNrrDorlLPL2-2
RidsRgn2MailCenter
KKarwoski, NRR/CSGB

ADAMS ACCESSION NUMBER: ML090750424 NRR-088

OFFICE	LPL2-2/PM	LPL2-2/LA	CSGB/BC	LPL2-2/BC
NAME	TOrf	RSola	AHiser By memo	TBoyce
DATE	3/17/09	3/17/09	2/3/09	3/17/09

OFFICIAL RECORD COPY