Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

March 16, 1998

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

10 CFR 50.54(f)

Gentlemen:

In the Matter of)
Tennessee Valley Authority)

Docket Nos. 50-327 50-390
50-328

SEQUOYAH NUCLEAR PLANT (SQN) AND WATTS BAR NUCLEAR PLANT (WBN) 90-DAY RESPONSE TO NRC GENERIC LETTER (GL) 97-05, "STEAM GENERATOR TUBE INSPECTION TECHNIQUES," DATED DECEMBER 17, 1997

This letter provides TVA's 90-day response to the subject GL. In accordance with the GL, TVA has performed the requested actions for SQN and WBN. This letter summarizes the actions taken, conclusions reached, and provides a summary description of the program that has been implemented for each site.

Enclosures 1 and 2 provide TVA's response for SQN and WBN, respectively.

If you have questions regarding this response, please contact Terry Knuettel at (423) 751-6673.

Sincerely,

Mark J. Burzynski

Mark J. Burzynski
Manager
Nuclear Licensing

D030

Subscribed and sworn to before me
this 16th day of March 1998

Paulette N. White
Notary Public

My Commission Expires 3-7-2001

ADD: Al Dimeglick

Enclosure
cc (Enclosure):
Mr. Mark S. Lesser, Chief
U.S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303

Mr. R. W. Hernan, Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

Mr. R. E. Martin, Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

NRC Senior Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy Daisy, Tennessee 37379

NRC Senior Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

TVA 90-DAY RESPONSE TO GENERIC LETTER (GL) 97-05,
"STEAM GENERATOR TUBE INSPECTION TECHNIQUES,"
DATED DECEMBER 17, 1997

NRC Required Action:

NRC's December 17, 1997, GL 97-05 requires addressees to submit a written response that includes the following information:

1. Whether it is the licensee's practice to leave steam generator tubes with indications in service based on sizing.

2. If the response to item (1) is affirmative, those licensees should submit a written report that includes, for each type of indication, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the techniques used.

TVA's Response

Introduction

GL 97-05, "Steam Generator Tube Inspection Techniques," was issued to (1) emphasize to the addressees the importance of performing steam generator tube in-service inspections using qualified techniques in accordance with the requirements of Appendix B to 10 CFR Part 50, and (2) require certain information from addressees to determine whether they are in compliance with the current licensing basis for their respective facilities given their steam generator tube in-service inspection practices.

This response provides information for SQN (Westinghouse Model 51 steam generators) requested by the GL. It is the practice at SQN to leave certain steam generator tubes with indications in service based on sizing if the indications are less than the 40 percent technical specification plugging limit. However, other than axial Outside Diameter Stress Corrosion Cracking (ODSCC) where a GL 95-05 alternate repair criteria applies, SQN does not leave crack-like indications in service. The information provided in this response includes for each type of indication that is left in service based on sizing, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the technique used.

The nuclear power industry recently voted to adopt an initiative requiring each utility to meet the intent of the guidance provided in Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines, no later than the first refueling outage starting after January 1, 1999. As required by NEI 97-06, each utility is required to follow the inspection guidelines contained in the latest revision of the Electric Power Research Institute (EPRI) PWR Steam Generator Examination Guidelines. The industry recommends implementation of Revision 5 to the guidelines by April 1, 1998.

PWR Steam Generator Examination Guidelines, Appendix H, "Performance Demonstration for Eddy Current Examination," Revisions 3 through 5 (depending on the timing of each units' refueling outage), provides guidance on the qualification of steam generator tubing examination techniques and equipment used to detect and size flaws. Damage mechanisms are divided into the following categories: thinning, pitting, wear, outside diameter intergranular attack/SCC, primary-side SCC, and impingement damage for qualification.

For qualification purposes, test samples are used to evaluate detection and sizing capabilities. While pulled tube samples are preferred, fabricated samples may be used. If fabricated test samples are used, the samples are verified to produce signals similar to those being observed in the field in terms of signal characteristics, signal amplitude, and signal-to-noise ratio. Samples are examined to determine the actual through wall defect measurements as part of the Appendix H qualification process.

The procedures developed in accordance with Appendix H specify the essential variables for each procedure. These essential variables are associated with an individual instrument, probe, cable, or particular on-site equipment configurations. Additionally, certain techniques have undergone testing and review to quantify sizing performance. The sizing data set includes the detection data set for the technique with additional requirements for number and composition of the grading units.

SIZING TECHNIQUES

At SQN, the following sizing techniques are used during steam generator inspections to leave flaws in service. The basis for application of these sizing techniques is the conduct of the examinations under the SQN Quality Assurance Program following the requirements of Sections XI and V of the ASME Code 1989 Edition and Regulatory Guide 1.83. Additional support for sizing degradation-specific mechanisms is provided by the EPRI Appendix H qualification data sets.

Wear

For wear at anti-vibration bars and flow lane blocking devices, sizing is accomplished using mixes [400/100 differential mix, 400/100 absolute mix, 200/100 absolute mix] of the bobbin probe. A calibration curve for amplitude vertical maximum is determined based on the applicable standards replicating the damage mechanism type and quantity. The calibration curve must represent the full range of expected depths.

This sizing qualification is based on 64 sample data points. The samples range in depth from 4 percent to 78 percent through wall. This database has been reviewed to ensure that application of the sizing procedure is consistent with the steam generator conditions at SQN. Therefore, the sizing procedure for wear is site-qualified for SQN in accordance with Section 6.2.4 of the PWR Steam Generator Examination Guidelines, Revision 5.

Thinning

For thinning at non-dented locations, the 400/100 kHz mix from the bobbin coil probe is used to size the extent of wall thinning. A calibration curve is established using the 20 percent, 60 percent, and 100 percent holes of the ASME calibration standard. The depth of the thinning is called from the 400/100 kHz mix using the maximum rate of change.

The sizing procedure is based on the analysis of 32 sample data points. Of the 32 data points, 5 are from pulled tubes. The samples ranged in depth from 9 percent to 100 percent. The thinning database has been reviewed to ensure that application of the sizing procedure is consistent with the steam generator conditions at SQN. Therefore, the sizing procedure for thinning is site-qualified for SQN in accordance with paragraph 6.2.4 of the PWR Steam Generator Examination Guidelines, Revision 5.

ODSCC at Non-dented Tube Support Plate Intersections

Sizing is performed in accordance with GL 95-05.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

TVA 90-DAY RESPONSE TO GENERIC LETTER (GL) 97-05,
"STEAM GENERATOR TUBE INSPECTION TECHNIQUES,"
DATED DECEMBER 17, 1997

NRC Required Action:

NRC's December 17, 1997, GL 97-05 requires addressees to submit a written response that includes the following information:

1. Whether it is the licensee's practice to leave steam generator tubes with indications in service based on sizing.

2. If the response to item (1) is affirmative, those licensees should submit a written report that includes, for each type of indication, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the techniques used.

TVA's Response

Introduction

GL Letter 97-05, "Steam Generator Tube Inspection Techniques," was issued to (1) emphasize to the addressees the importance of performing steam generator tube in-service inspections using qualified techniques in accordance with the requirements of Appendix B to 10 CFR Part 50, and (2) require certain information from addressees to determine whether they are in compliance with the current licensing basis for their respective facilities given their steam generator tube in-service inspection practices.

This response provides information for WBN (Westinghouse Model D-3 steam generators) requested by the GL. It is the practice at WBN to leave certain steam generator tubes with indications in service based on sizing if the indications are less than the 40 percent technical specification plugging limit. However, WBN does not leave crack-like indications in-service. The information provided in this response includes for each type of indication that is left in service based on sizing, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the technique used.

The nuclear power industry recently voted to adopt an initiative requiring each utility to meet the intent of the guidance provided in Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines, no later than the first refueling outage starting after January 1, 1999. As required by NEI 97-06, each utility is required to follow the inspection guidelines contained in the latest revision of the Electric Power Research Institute (EPRI) PWR Steam Generator Examination Guidelines. The industry recommends implementation of Revision 5 to the guidelines by April 1, 1998.

PWR Steam Generator Examination Guidelines, Appendix H, "Performance Demonstration for Eddy Current Examination," Revisions 3 through 5, provides guidance on the qualification of steam generator tubing examination techniques and equipment used to detect and size flaws. Damage mechanisms are divided into the following categories: thinning, pitting, wear, outside diameter intergranular attack/SCC, primary-side SCC, and impingement damage for qualification.

For qualification purposes, test samples are used to evaluate detection and sizing capabilities. While pulled tube samples are preferred, fabricated samples may be used. If fabricated test samples are used, the samples are verified to produce signals similar to those being observed in the field in terms of signal characteristics, signal amplitude, and signal-to-noise ratio. Samples are examined to determine the actual through wall defect measurements as part of the Appendix H qualification process.

The procedures developed in accordance with Appendix H specify the essential variables for each procedure. These essential variables are associated with an individual instrument, probe, cable, or particular on-site equipment configurations. Additionally, certain techniques have undergone testing and review to quantify sizing performance. The sizing data set includes the detection data set for the technique with additional requirements for number and composition of the grading units.

SIZING TECHNIQUES

At WBN, the following sizing techniques are used during steam generator inspections to leave flaws in service. The basis for application of these sizing techniques is the conduct of the examinations under the WBN Quality Assurance Program following the requirements of Sections XI and V of the ASME Code 1989 Edition and Regulatory Guide 1.83. Additional support for sizing degradation-specific mechanisms is provided by the EPRI Appendix H qualification data sets.

Wear

For wear at anti-vibration bars and tube support plates, sizing is accomplished using the 540/130 absolute mix of the bobbin probe. A calibration curve for amplitude vertical maximum is determined based on the applicable standards replicating the damage mechanism type and quantity. The calibration curve must represent the full range of expected depths.

This sizing qualification is based on 64 sample data points. The samples ranged in depth from 4 percent to 78 percent through wall depth. This database has been reviewed to ensure that application of the sizing procedure is consistent with the steam generator conditions at WBN. Therefore, the sizing procedure for wear is site-qualified for WBN in accordance with paragraph 6.2.4 of the PWR Steam Generator Examination Guidelines, Revision 5.

Potential Loose Parts Damage

During the first refueling outage at WBN, 15 indications were detected in SG 1 in 14 tubes at varying elevations from 0.5 inches to 14 inches between the fifth and sixth support plate on the cold leg immediately in front of the

feedwater nozzle inlet. Some of the indications were just above the fifth support plate and appeared by eddy current to be the result of loose parts damage. There were others several inches above the support plate which appeared to be localized loose parts damage. It is not unusual for plants to see loose parts damage in the first outage inspection. Also, WBN had a number of upper internals and preheater modifications after the generators were installed. For these reasons, loose parts damage was predicted in the preoutage damage assessment.

Each indication was evaluated with the +Point coil to further characterize the signal. Many of the signals exhibited small volume. None of the responses from the +Point tests appeared crack-like. Since the indications were similar in eddy current response to the ASME standard, depths were assigned to these indications based on bobbin coil phase analysis at 550 kHz differential. Indications of wear were not detected within the thickness of the tube support plate. The tubes affected were in groupings of common rows or columns. The maximum depth indication of this type was reported as 75 percent through wall (R47 C64) by bobbin. Based on low frequency (10 KHz) rotating probe coil inspection, a suspected metallic object was detected adjacent to one of the tubes (R49 C57) with a wear indication. The characterization of these indications was fairly uniform among the tubes involved. Of the 15 total indications, 3 exceeded the 40 percent depth based plugging limit and were removed from service. Based on the identification of the suspected foreign object, tube R49 C57 was preventively plugged.

The 75 percent maximum depth indication located in R47 C64 was at approximately 6 inches above the fifth cold leg plate. As the tube vibration amplitudes increase at elevations approaching the midspan, it is reasonable that largest depth indications were located away from the plate. This bounding indication had insufficient volume to present a rupture or leakage concern.

Loose parts damage was also identified in SG 4 in periphery tubes R13 C3 (40 percent through wall) and R14 C3 (25 percent through wall). R13 C3 was taken out of service. Neither indication represented rupture or leakage concerns.

Inspections bound the affected areas, periphery tubes, and the preheater regions. Since no leakage occurred from the 75 percent indication in R47 C64, its size presented no rupture concern, and 100 percent of the potentially affected areas were inspected, it can be concluded that no indication was left undetected at the end-of-cycle 1 that would present a rupture or leakage concern. Since tubes with loose parts damage are not in a continued state of degradation, growth should not be projected in the tubes left in-service with less than 40 percent wall loss.