February 18, 2004

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

Gentlemen:

In the Matter	)	Docket No. 50-328
Tennessee Valley Authority	)	

SEQUOYAH NUCLEAR PLANT (SQN) - UNIT 2 – RESPONSE TO REQUEST FOR INFORMATION REGARDING STEAM GENERATOR (SG) INSPECTION REPORTS FROM UNIT 2 CYCLE 11

References: 1. TVA letter to NRC dated May 5, 2003, "Sequoyah Nuclear Plant – Unit 2 Cycle 11 (U2C11) 12-Month Steam Generator (SG) Inspection Report and Metallurgical Examination Report on Tube Removed from SG"

2. TVA letter to NRC dated July 29, 2002, "Sequoyah Nuclear Plant (SQN) – Unit 2 – Unit 2 Cycle 11 (U2C11) 90-Day Steam Generator Report For Voltage-Based Alternate Repair Criteria"

Enclosed are TVA's responses to NRC staff questions regarding the reference 1 and 2 SG reports. As discussed with the staff, the additional questions were received on September 30, 2003, regarding the SQN Unit 2 Cycle 11 SG Inspection Reports.

Enclosure 1 provides responses to staff questions associated with Reference 1. Enclosure 2 provides responses to staff questions associated with Reference 2. Enclosure 3 provides responses to staff questions associated with References 1 and 2, including TVA's May 8, 2002 Tube Plugging Report and the September 17, 2002 NRC Outage Conference Call Summary. Enclosure 4 provides a list of associated references.

There are no commitments contained in this submittal. Please direct questions concerning this issue to me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Original signed by:

Pedro Salas
Licensing and Industry Affairs Manager

Enclosures

ENCLOSURE 1

SEQUOYAH NUCLEAR PLANT

NRC QUESTIONS AND TVA RESPONSES

UNIT 2 CYCLE 11 STEAM GENERATOR INSPECTIONS

The following questions and responses are based on TVA's letter to NRC dated May 5, 2003, Enclosure 2 "Final Metallurgical Report For Steam Generator Tube R12C45."

NRC Question 1.

On pages 22-23, the report, indicates that during the burst test of section 3B (TSP1), the specimen was semi-restrained by a simulated support system designed to mock the conditions in the Sequoyah-2 SGs under accident conditions. The centerline of the TSP1 region on section 3B was positioned 2 inches above the centerline of the support plate simulation. Discuss whether the presence of the simulated support plate influenced the burst test results (e.g., increased the resulting burst pressure, etc.). In addition, explain the purpose in semi-restraining the specimen for the burst test considering the tests are supposed to be performed under assumed freespan conditions.

TVA Response

The Westinghouse pulled-tube burst tests are performed to simulate free span conditions with lateral constraint. Lateral constraint is necessary to obtain an acceptable influence of circumferential degradation on burst pressure. Since some axial outside diameter stress corrosion cracking (ODSCC) indications may have significant cellular corrosion, the burst tests are generally performed with lateral constraint to assure that the appropriate influence of the cellular corrosion is included in the burst test. Prior to the burst test, there is no acceptable method for estimating potential cellular corrosion so most pulled-tube burst tests include lateral constraint. The tube support plate (TSP) simulant located 47-48 inches above the TSP crevice location of the specimen to be burst provides lateral constraint in the burst tests and includes nominal tube to TSP clearances. The specimen to be burst is supported by a split collar (negligible clearances) such that the bottom of the TSP crevice is about 2 inches above the top of the collar. A tube extension is added to the burst specimen to extend up to the TSP simulant. The same general arrangement can be used for top of tubesheet (TTS) specimens by locating the split collar at the TTS location on the burst specimen.

The influence of split-collar support on burst pressure falls off rapidly with the distance between the top of the crack and the support. For example, the Electric Power Research Institute (EPRI) primary water stress corrosion cracking (PWSCC) alternate repair criteria (ARC) development for hardroll transitions (Section 4.5 of Reference 7) specifies that the axial length reduction for tubesheet constraint should be assumed to be zero if the lower crack tip does not extend to the tube to tubesheet contact point (bottom of expansion transition) or if the upper crack tip is more than 22 millimeters (0.87 inches) from the tubesheet contact point. When the distance from the bottom of the crack to the tubesheet contact point exceeds about half an inch, the influence of the tubesheet constraint on the burst pressure can be neglected. For the pulled-tube burst tests with the bottom of the flaw at least 2 inches above the split collar, there would be no influence of the collar on the burst pressure. It can be concluded that the presence of the TSP simulant and split collar have no influence on the burst pressure of the axial crack two inches above the collar other than bending restraint, and the measured burst tests are properly classified as free span bursts.

NRC Question 2

The report indicates that the indications in the top of tubesheet (TTS) region appeared shallow based on the destructive examinations. Describe the sizing of these flaws in more detail. of the eddy current inspection technique, which did not detect this indication.

TVA Response

The TTS indications are a Westinghouse Explosive Tube Expansion (WEXTEX) expansion transition axial ODSCC indication that extends slightly above the top of the transition located at about the top of the tubesheet. The crack position is influenced by deposits within and above the top of the tubesheet (i.e., above the bottom of the expansion transition) such that the crack morphology of multiple axial cracks is similar to that of a sludge pile or TSP indication. The specimen burst at 11,453 pounds per square inch (psi) or only 300 psi below that of a non-degraded tube section, thus indicating a shallow or short crack. The results of the fractography of the burst opening indicated that the greater than

3 percent through-wall portion of the crack was approximately 0.21 inches in length with another approximately 0.07 inches long tail with a through wall depth of 3 percent or less. The reported local maximum depth for the burst crack is 43 percent but the maximum depth running average over 0.1 inch is only 30 percent. A transverse circumferential section cut near the top of the tubesheet identified a crack with a depth of 49 percent but this crack (length cannot be determined from a transverse section) must have been too short to influence the burst pressure.

Since the indications are dominantly WEXTEX expansion transition cracks, detection by a bobbin coil would not be expected due to the influence of the expansion on coil response. The TTS indication was not reported in either the field or post-pull laboratory +Point inspection. The laboratory pancake coil inspection detected a 0.8 volt indication, but this indication was associated with the TIG process applied during the tube removal operations. The combination of the short length with a shallow effective maximum depth of 30 percent in conjunction with the effects of the expansion transition and deposits made these cracks difficult to detect.

ENCLOSURE 2

SEQUOYAH NUCLEAR PLANT

NRC QUESTIONS AND TVA RESPONSES

UNIT 2 CYCLE 11 STEAM GENERATOR INSPECTIONS

The following questions and responses are based on TVA's letter to NRC dated July 29, 2002, Enclosure 1 "Sequoyah Nuclear Plant Unit 2 Cycle 11 Refuel Outage Condition Monitoring and Operational Assessment: GL 95-05 Voltage Based Alternate Repair Criterion End Of Cycle 11-90 Day"

Preface

Calculations were performed to compare probability of burst and leakage rate at end of cycle (EOC)-11 with the Generic Letter (GL) 95-05 acceptance criteria and the predictions performed at the previous inspection. The predictions computed for EOC-11 were reported in the EOC-10, 90-Day Report dated January 2001 (Reference 1). These predictions were based on the Addendum 3 database (Reference 2), as that was the most current data at that time. The calculations performed at the EOC-11 were based on the Addendum 4 database (Reference 3) updated to include data from Beaver Valley Unit 1 (References 4 and 5). Because the Beaver Valley data significantly affected the leakage rate, a revised prediction for EOC-11 leakage based on EOC-10 voltage distribution and updated correlations was made and reported in Table 6.1 of the subject report. The probability of burst was not revised because it was not affected significantly. The revised predicted leakage rates were found to be conservative when compared to the leakage computed for condition monitoring using the EOC-11 voltage distribution. The probability of burst for steam generator (SG) 4 was greater than predicted by a small amount, and that will be discussed in response to Questions 3 and 4.

NRC Question 1

Discuss how the condition monitoring and operational assessment results were affected, as the results relate to the performance criteria (i.e., probability of burst (POB) of 1x10-2 and leakage of 8.2 gallons per minute (gpm)), by the tube pull leak and burst test results discussed in the May 5, 2003 letter to the NRC.

TVA Response

No specific analysis has been performed to address the impact of the pulled tube data. The Addendum 5 database (Reference 5) includes the data from Sequoyah Unit 2 as well as Beaver Valley Unit 1. Other modifications to the database included removal of French data as approved by NRC. In order to evaluate the sensitivity of the results to the change in database, analyses were performed on the one Sequoyah Unit 2 SG with the highest predicted leakage, SG 4.

Monte Carlo Condition Monitoring Analysis Results for Measured EOC-11 Voltage Distribution, Steam Generator 4

Analysis Basis	Number of Trials	Number of Indications	Max Volts Measured	Number of bursts	Burst Probability 95% Conf.	SLB Leak Rate, gpm 95/95
Subject Report	250,000	621	3.35	9	6.3 x 10^-5	1.29
ADD. 5 database	250,000	621	3.35	24	1.35 x 10^-4	0.701

The Addendum 5 database causes the POB to increase (by a factor of 2), but the leakage rate decreases. These trends were anticipated in Reference 5.

Monte Carlo Operational Assessment Analysis Results for EOC-12 Voltage Distribution, Steam Generator 4

Analysis Basis	Number of Trials	Number of Indications	Max Volts Measured	Number of bursts	Burst Probability 95% Conf.	SLB Leak Rate, gpm 95/95
Subject Report	250,000	1024	3.7	11	7.3 x 10^-5	3.40
ADD. 5 database	250,000	1024	3.7	52	2.62 x 10^-4	2.06

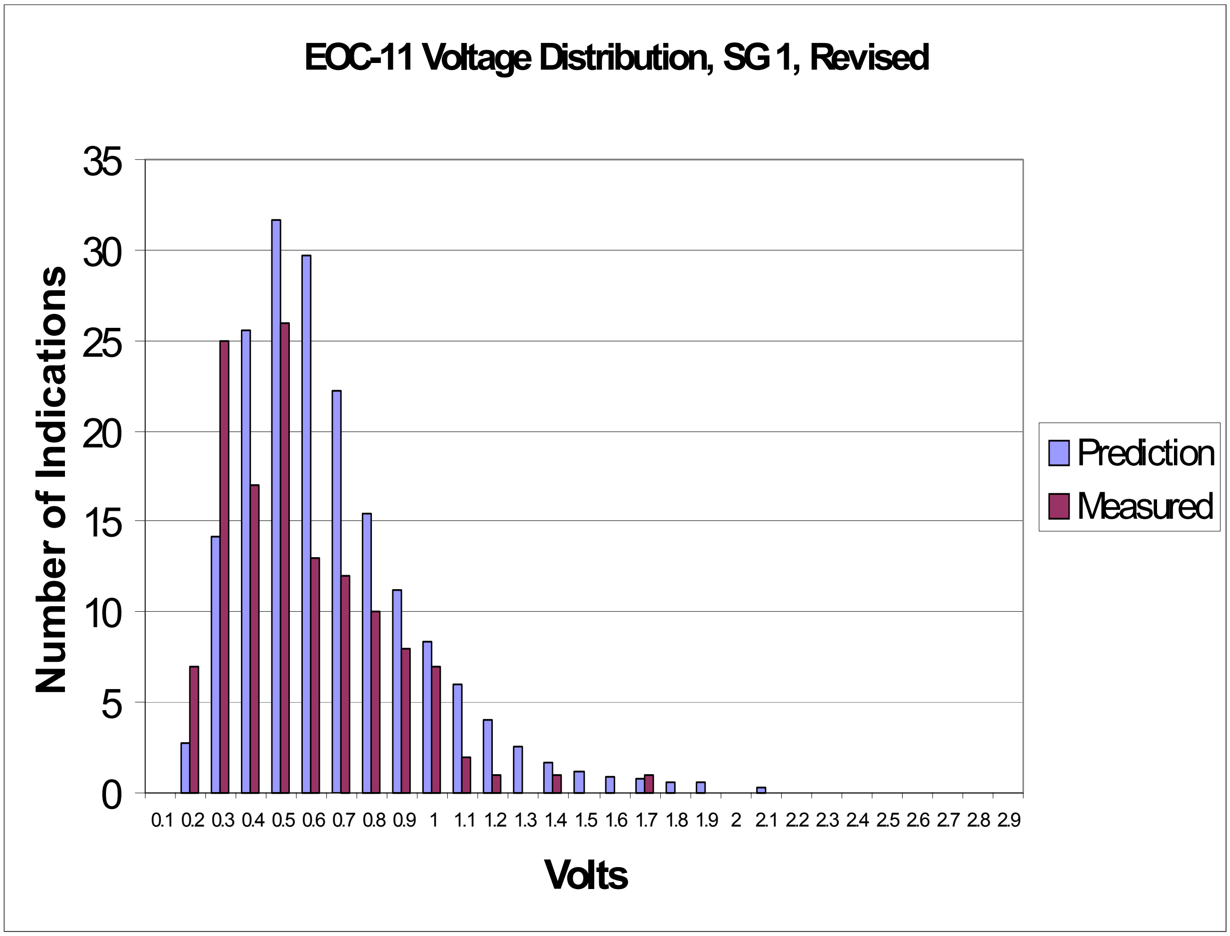
Again, the Addendum 5 database causes the POB to increase, but the leakage rate decreases.

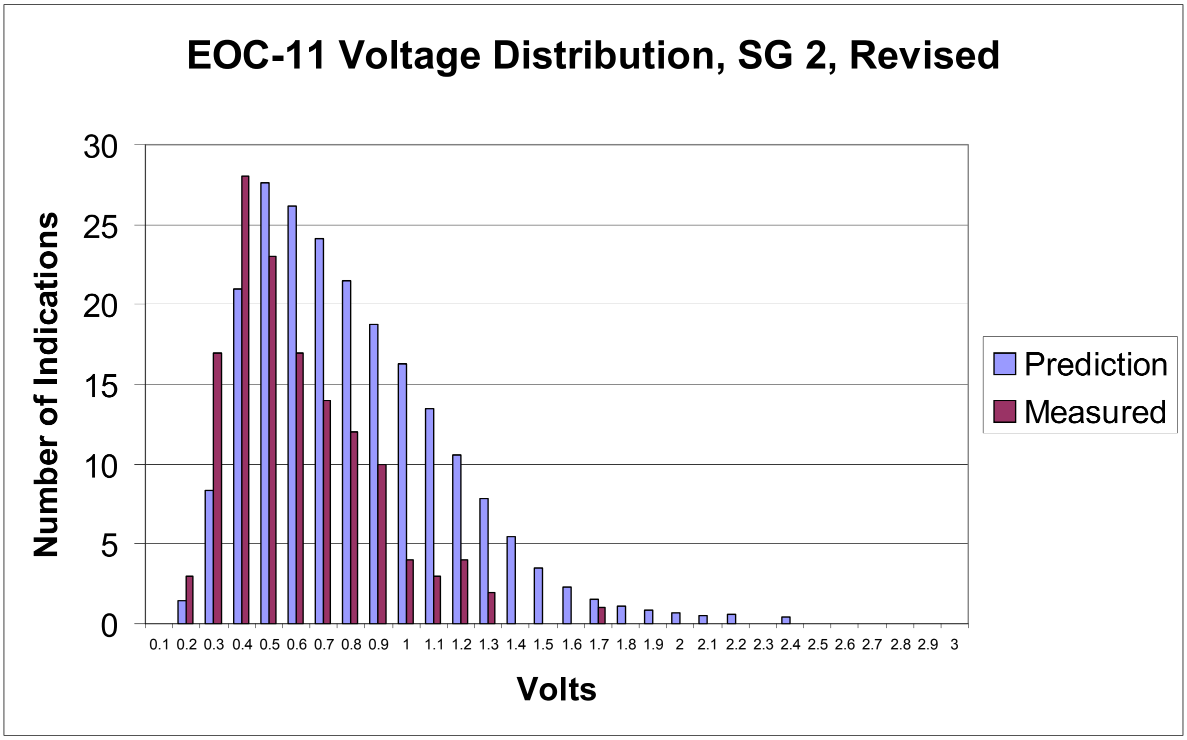
NRC Question 2

The EOC-11 voltage distribution predictions identified in Figures 4.1 - 4.4 of the July 29, 2002 report appear to be different than the EOC-11 voltage distribution predictions identified in Figures 7.1 - 7.4 of the January 30, 2001 report submitted to the NRC. Explain this apparent discrepancy. If errors are present in one or both sets of figures, clarify if correction of the errors changes any of the details (e.g., results or comparisons) in either of the two reports.

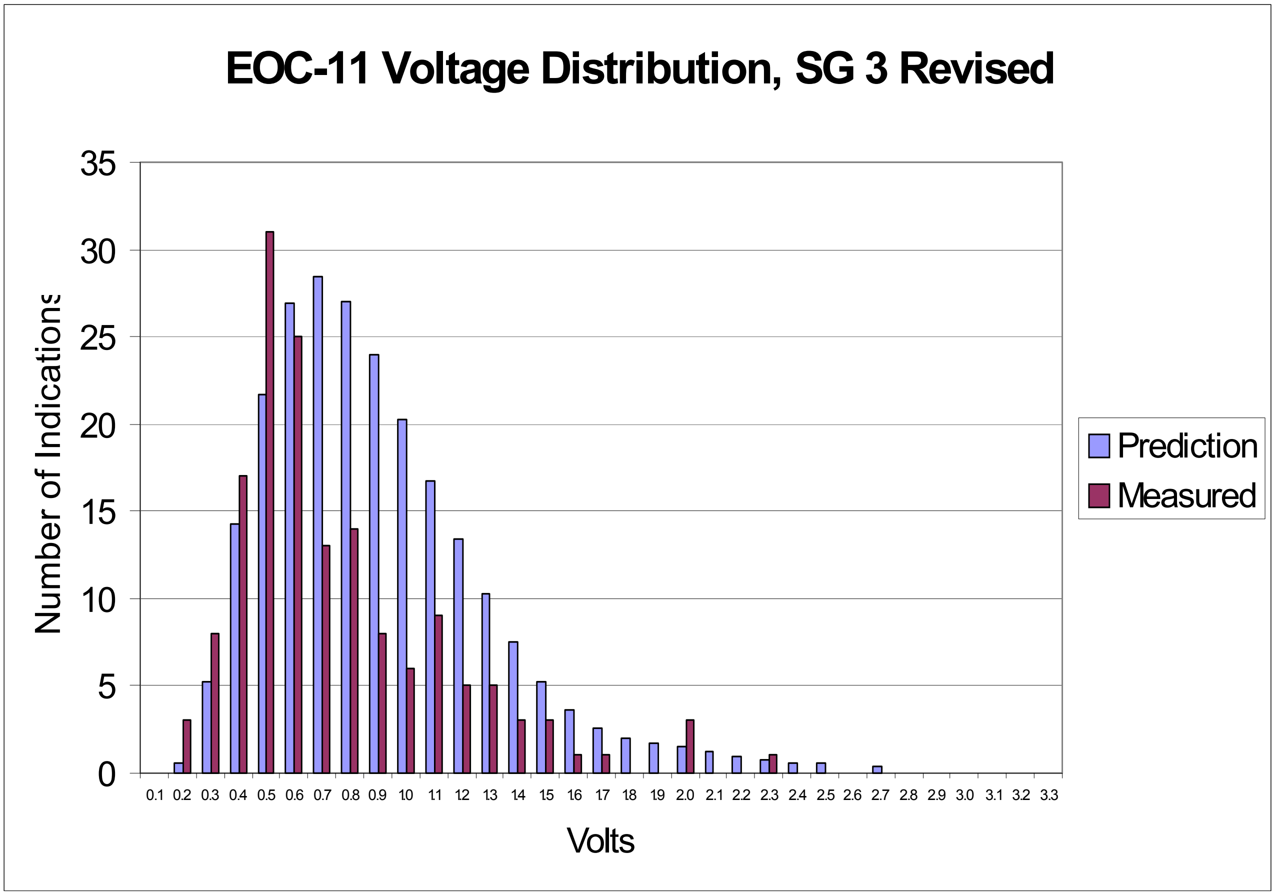
TVA Response

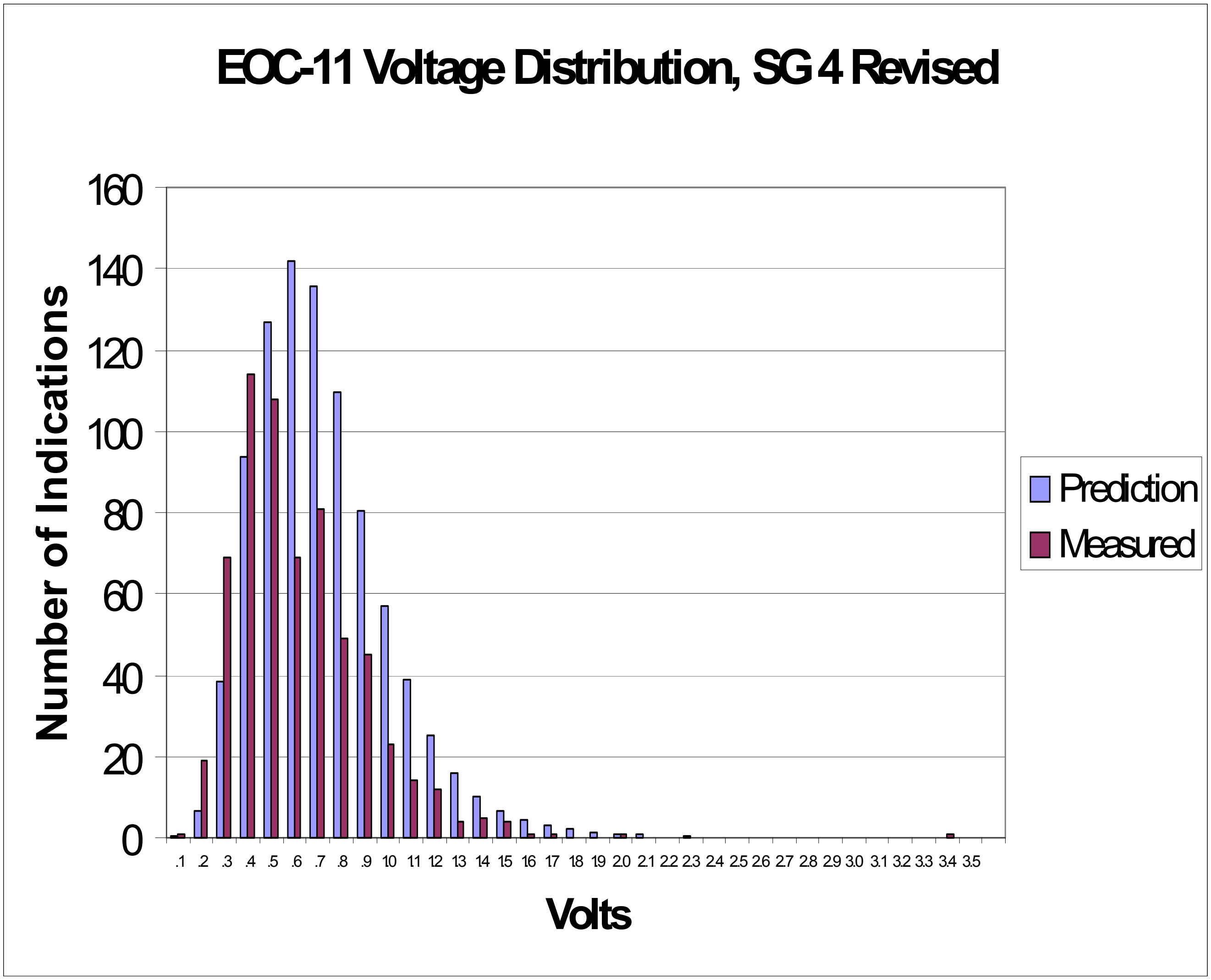
A clerical error was made by the author in the preparation of the final figures of the 2002 report. All comparisons were made with the correct data. This error resulted from a final "clean-up" of the figures. We regret the error and have corrected the figures via TVA's Corrective Action Program (Reference SQN PER 04-770190). The following corrected figures are provided.


EOC-11 Voltage Distribution, SG 1, Revised
35
30
25
20
15
10
5
0
Number of Indications
0.1 0.2 0.3 0.4 0.5 0.6 0.7 0.8 0.9 1 1.1 1.2 1.3 1.4 1.5 1.6 1.7 1.8 1.9 2 2.1 2.2 2.3 2.4 2.5 2.6 2.7 2.8 2.9
Volts
Prediction
Measured



EOC-11 Voltage Distribution, SG 2, Revised
Number of Indications
0
5
10
15
20
25
30
0.1 0.2 0.3 0.4 0.5 0.6 0.7 0.8 0.9 1 1.1 1.2 1.3 1.4 1.5 1.6 1.7 1.8 1.9 2 2.1 2.2 2.3 2.4 2.5 2.6 2.7 2.8 2.9 3
Volts
Prediction
Measured


EOC-11 Voltage Distribution, SG 3 Revised
Number of Indications
35
30
25
20
15
10
5
0
0.1 0.2 0.3 0.4 0.5 0.6 0.7 0.8 0.9 10 11 12 13 14 15 16 17 18 19 2.0 2.1 2.2 2.3 2.4 2.5 2.6 2.7 2.8 2.9 3.0 3.1 3.2 3.3
Volts
Prediction
Measured


EOC-11 Voltage Distribution, SG 4 Revised
Number of Indications
160
140
120
100
80
60
40
20
0
.1 .2 .3 .4 .5 .6 .7 .8 .9 1.0 1.1 1.2 1.3 1.4 1.5 1.6 1.7 1.8 1.9 2.0 2.1 2.2 2.3 2.4 2.5 2.6 2.7 2.8 2.9 3.0 3.1 3.2 3.3 3.4 3.5
Volts
Prediction
Measured

NRC Question 3

For SG 4, the EOC-11 predicted POB (1.82E-5) was nonconservative as compared against the actual EOC-11 POB (6.3E-5) based on the as-found distribution of indications. The report states that the actual EOC-11 POB for SG 4 is higher than the prediction because the maximum voltage indication measured was greater than the maximum predicted. Clarify if this is intended to imply that the predicted POB would have been conservative if the 3.35 volt indication (the largest voltage indication) was not present. If this is not the intent, provide a more robust explanation for the underprediction.

TVA Response

The following is a more detailed explanation of the underprediction:

The predicted probability of burst (POB) of 1.82 E-5 was based on Addendum 3 data and 95 percent confidence results with 1,000,000 Monte Carlo trials. The actual (condition monitoring) POB of 6.3 E-5 was based on Addendum 4 data (modified to include Beaver Valley tube pull data) and 95 percent confidence with 250,000 Monte Carlo trials. In order to assess the effect of these differences, analyses were made using both data sets and the same number of Monte Carlo trials. Using the Addendum 4 data (modified to include Beaver Valley tube pull data) and 1,000,000 trials, the condition monitoring POB is 3.26 E-5, closer to the predicted value. Using the Addendum 3 data and 1,000,000 trials, the condition monitoring POB is 2.78 E-5, indicating that the database used makes a very small difference. The dominant cause of the difference in results is the number of Monte Carlo trials. Given the same number of Monte Carlo trials, the difference between the predicted and actual POB at EOC-11 is less than a factor of two. It is clear that the differences in results are small and not significant relative to the differences that can be obtained by changing some analysis parameters (such as number of trials, Monte Carlo seed, etc.). The significance of the difference between a POB of 1.82 E-5 and 6.3 E-5 is further discussed in the response to question 4.

NRC Question 4

With regards to the issue in the previous question, the report states that despite the nonconservative nature of the POB prediction for SG4, all steam generators were well below the burst acceptance criterion of 1.0E-2 and that the acceptance criterion on POB are satisfied with significant margin. The Nuclear Regulatory Commission (NRC) staff recognizes that from a safety perspective, differences between the predicted POB and actual POB are only significant when the acceptance criterion are being approached. However, to ensure that trends that may be indicative of a non-

conservative methodology are promptly detected, a more appropriate measure for assessing the adequacy of a methodology may be beneficial (such as when the projections and actual results differ by some specified value [e.g., 10 percent]). Discuss any plans to change the methodology if this trend continues.

TVA Response

As noted in the response to Question 3, the differences between the POB projections and as-measured, when performed with the same data base and number of Monte Carlo simulations, is less than a factor of two with a magnitude of about 3 E-5. This difference is negligible.

Differences between projections and actual results on the order of 10 percent are much too small to consider methods changes unless the magnitudes are near the acceptance levels. More acceptable thresholds for assessing the adequacy of the methods would be the values identified in Section 3.4 of the Diablo Canyon Probability of Prior Cycle Detection (POPCD) safety evaluation report (SER) (Reference 6). Per this reference, a significant underprediction of burst probability is defined as 10 percent of the reporting threshold (i.e., 0.001). A significant underprediction of Steam Line Break (SLB) leak rate is defined as 0.5 gpm. For smaller burst probabilities or leak rates, a methods assessment would be performed if the condition monitoring results are underpredicted by an order of magnitude.

NRC Question 5

Given that the underprediction of the POB did occur, what actions, if any, were taken this cycle to prevent recurrence?

TVA Response

As noted in the responses to Questions 3 and 4, the POB underprediction is negligible in magnitude and largely attributable to sensitivity to the number of Monte Carlo samples when trying to predict burst probabilities on the order of E-5. No actions are necessary or were taken to prevent recurrence of these small differences as noted in the response to Question 4.

NRC Question 6

The number of small voltage indications (ranging from 0.1 - 0.5 volts) were underpredicted for all four steam generators. Consideration should be given for developing an approach which increases the number of indications to account for this phenomena. That is, if the initiation rate is increasing, some sort of an "acceleration" factor should be considered. Discuss any plans to change the methodology if this trend continues.

TVA Response

The EOC-11 projections summed over all SGs underestimate the as measured population between 0.0 and 0.4 volts by less than 100 indications while overestimating the total population by more than 480 indications. For a single SG, SG 4 underestimates the <0.4 volt population by about 75 indications while overestimating the total population by about 274 indications. In terms of the number of indications, the projections are extremely conservative with substantial overestimates of the higher voltage population. The differences between projections and measurements can be clearly seen in the figures provided in the response to Question 2.

The principal contributor to the differences between projections and actuals is the GL 95-05 requirement to apply a constant POD of 0.6. This POD is too high below 0.6 volts, which leads to underestimates of the low voltage population. Since a probability of detection (POD) of 0.6 is much too low above 0.6 volts, the analyses overestimate the higher voltage population. The appropriate methodology change to address these differences would be to apply a voltage dependent POD, such as POPCD. Since POPCD requires NRC approval, it cannot be implemented at this time. There is no need or plans to change the methodology for underestimates of the low voltage population when the total population is adequately or conservatively predicted such as obtained at EOC-11. This conclusion is consistent with the Diablo Canyon POPCD SER (Reference 6) which recommends a methods assessment when the total number of as-found indications is underestimated by >15 percent.

NRC Question 7

The NRC staff recently became aware that some licensees implementing the Generic Letter (GL) 95-05 voltage based repair criteria were addressing intersections with large mixed residuals differently than described in GL 95-05 (Attachment 1, Sections 1.b.3 and 3.b.4). Describe the methodology and process through which you are addressing this issue and the basis for your actions. [Refer to ADAMS Document ML031550196, Enclosure 3 for background information on this issue if needed.]

TVA Response

For the application of the alternate repair criteria for ODSCC at tube support plates, TVA has implemented controls to address issues of non-flaw bobbin probe signals masking ODSCC indications of 1 volt or greater.

In the analysis process at TVA, data is evaluated for quality by data quality analysts prior to being sent to the analysts as well as by each analyst evaluating the data for indications. Each analyst is tasked with the responsibility of identifying data which could mask indications. The following excerpt taken from the guidelines points out the emphasis with which the importance of analysts evaluating areas where indications might be masked, "It is the responsibility of all personnel involved with the analysis process to identify conditions which inhibit the evaluation of the data."

For detection of mixed residual that could mask a 1-volt ODSCC indication, TVA utilizes a computerized data screening (CDS) system with the bobbin probe to identify potential mixed residual indications. This screen is set to a conservative level (1.6 volts) based on a review of SQN data. This screen identifies support plates to be reviewed by the resolution analysts. These analysts are the more experienced analysts who are familiar with TVA data and methods. The resolution analysts review the support plates identified by CDS to determine if the potential exists for a 1 volt ODSCC indication to be masked by mixed residual at the support plate. If the analyst believes that the mixed residual could mask a 1-volt ODSCC indication, a mixed residual indication (MRI) is placed in the indication field and that indication is examined by a technique such as the rotating +Point probe. If the signal identified by bobbin is confirmed by +Point the tube is repaired.

ENCLOSURE 3

SEQUOYAH NUCLEAR PLANT

NRC QUESTIONS AND TVA RESPONSES

UNIT 2 CYCLE 11 STEAM GENERATOR INSPECTIONS

The following questions are miscellaneous questions not related to Generic Letter 95-05. The sources of information for these questions and responses include TVA letters to NRC dated July 29, 2002, May 5, 2003, May 8, 2002 (Tube Plugging Report), and the September 17, 2002 NRC Outage Conference Call Summary (ADAMS ACCESSION #ML022600174).

NRC Question 1

The NRC staff notes that the inspection of the low row U-bend region in the Sequoyah Unit 2 SGs with a rotating probe during the previous outage consisted of 100% of Row 1- 3 tubes and 20% of Row 4 tubes in all four steam generators. Recent (Spring 2003) operating experience indicates that higher row U-bends may also be susceptible to PWSCC. Discuss whether any inspections (other than bobbin coil inspections) were performed in the U-bend region of higher row tubes during the Spring 2002 refueling outage. In addition, discuss whether any testing or analysis has been performed which indicates that the U-bend region of the lower row tubes will always exhibit degradation before the U-bend region of higher row tubes (consideration should be given to all factors that affect tube degradation including material properties and stresses). Lastly, discuss the inspection plans for the higher row U-bend regions with a rotating probe for the upcoming outage at Sequoyah Unit 2.

TVA Response

SQN did not test higher row U-Bends with probes other than bobbin in the U2C11 inspection. However, since the recent experience at Diablo Canyon, TVA initiated a Westinghouse Owners Group task to address the issue of U-Bend stresses.

Westinghouse prepared a report through the Westinghouse Owners Group to document and categorize the residual stresses in U-Bends for Westinghouse Model 51 SGs, to relate the conclusions to inspection results at Diablo Canyon and Beaver Valley, and to formulate inspection recommendations for U-Bends. As a result of this report, TVA inspected 100 percent of U-Bends in

Rows 1 through 11 and 20 percent sample of Rows 12 through 20. TVA used the Mitsubishi Intelligent Array (MHI)probe for this inspection except for the lower rows where the MHI probe is too large to traverse the bend. In the lower rows, TVA used a +Point probe.

NRC Question 2

Several reports indicate that an axial outside diameter stress corrosion cracking (ODSCC) indication was identified in a 2.65 volt freespan ding. The NRC staff understands the inspection scope was increased from: the original scope of rotating probe inspection of 20% of freespan dings greater than 2 volts from the hot leg TTS to the second hot leg tube support plate, to the expanded scope of rotating probe inspection of 100% of the hot leg freespan dings greater than 5 volts. Discuss whether the axial indication in the 2.65 volt freespan ding was detected by the bobbin probe, or whether it was only detected by the rotating probe. If this indication was not detected with the bobbin probe, discuss the basis for limiting the expansion to dings greater than 5 volts.

TVA Response

The indication in question is approximately one inch from the top of the tubesheet. It is believed that the ding occurred due to a sludge lance cart hitting the tube. This ding became the stress riser for the initiation of the indication. The indication was identified by both +Point analysis and bobbin analysis and is clearly seen in the eddy current lissajous presentation. TVA trains the analysts on the potential of ODSCC occurring at dents as was seen at South Texas, and utilizes the technique that was qualified by South Texas for freespan dings less than or equal to 5 volts.

NRC Question 3

Page 7 of Enclosure 1 of the May 5, 2003 report indicates that four indications of cold leg thinning exceeded the 40% throughwall repair limit and were plugged. Attachment 2 of the same enclosure indicates that eleven tubes were plugged due to cold leg wastage, but does not identify any tubes that were plugged due to thinning.

Clarify how many tubes were identified to contain indications of cold leg thinning. Specify how many of these tubes were plugged and the basis. Identify the location within the SG where indications of cold leg thinning were identified.

Clarify how many tubes were identified to contain indications of wastage. Specify how many of these tubes were plugged and the basis. Identify the location within the SG where indications of wastage were identified.

Describe the methodology used to differentiate between indications of cold leg thinning and wastage.

Operating experience indicates that wastage typically occurs in plants under phosphate water treatment conditions. Discuss whether Sequoyah Unit 2 has ever operated under phosphate water treatment conditions. If not, discuss the driving mechanism for the wastage occurring at Sequoyah Unit 2.

<u>TVA Response</u>

Within the subject report, TVA used cold leg thinning and cold leg wastage interchangeably. The damage mechanism for SQN Unit 2 is more correctly characterized as cold leg thinning. SQN Unit 2 started up with all volatile treatment (AVT) chemistry and has never utilized phosphate water treatment to control secondary side chemistry. Eleven tubes were identified in Enclosure 1 of the subject report and only four tubes characterized with cold leg thinning exceeded the 40 percent through-wall plugging criteria; the other 7 tubes were plugged preventively.

Attached is a list of cold leg thinning indications from Unit 2 Cycle 11. The "IND" column is the eddy current indication call. If a number is in the field, this is a percent wall loss call by bobbin coil. These indications are plugged using a 40 percent repair limit.

If an "SVI" is in the "IND" field, this is an indication that was inspected with +Point to ensure that the indication was not crack-like. These indications were confirmed to be volumetric indications, characterized as cold leg thinning, and were plugged preventively. More discussion on the characterization process is in the response to Question 4.

<u>NRC Question 4</u>

In Enclosure 3 of the July 29, 2002, report, a summary is provided which describes the basis for determining the presence of cold leg thinning versus axial ODSCC at cold leg tube support plates.

The report states that indications which are inspected with a rotating probe result in different eddy current signal

responses, dependent on whether the indication is volumetric in nature (i.e., thinning) or axial in nature (i.e., axial ODSCC). Given that predominantly axial ODSCC at tube support plate intersections can consist of a large number of closely spaced axial ODSCC that may appear volumetric in the rotating probe eddy current data, discuss how these conditions would be distinguished from thinning.

The report states that indications that are not distorted in the bobbin probe data are not inspected with a rotating probe and are assumed to be caused by cold leg thinning. Please discuss the technical basis for concluding that a non-distorted bobbin indication is the result of cold leg thinning versus some other form of degradation including wear, wear due to loose parts, and/or a large crack. Provide a summary of any qualification program that demonstrates the ability to distinguish degradation mechanisms based on bobbin probe signals.

TVA Response

TVA has examined hundreds of axial indications with +Point in the hot legs of the SGs and has no instances of axial indications appearing volumetric in nature. TVA follows the ODSCC from its detectable stage until it has fully matured. During this time, multiple or large macro cracks may appear, but TVA has not seen ODSCC become volumetric in nature. Cold leg indications are scrutinized as to their eddy current signal characteristics, location within the bundle, history, and the type of inspection technique being utilized. Each of these is used to distinguish thinning from cracking.

SQN Unit 2 is a Model 51 Westinghouse recirculating SG. Prior history of the Model 51s has revealed a potential for cold leg thinning. The areas of the SG most susceptible have been documented in EPRI, Steam Generator Reference Book, TR-103824, and Chapter 16 discusses cold leg thinning. The signal characteristics from cold leg thinning are different than what is seen in wear indications. Also, support plate wear would most likely occur at one or both edges of the support while cold leg thinning will be nearer the center of the support plate.

TVA uses bobbin coil ETSS 96001.1 to detect and size these indications. Distorted indications are tested with +Point. Those indications which are confirmed by the +Point test typically show round shaped flaws confined within the tube support. The plus point coil produces a bi-polar signal response which is typical for a volumetric flaw. This is distinctly different from ODSCC which produces a response only on the coil leg sensitive to axial flaws. Bobbin indications which are not distorted, which have signal characteristics like cold leg thinning, and which yield depth estimates of less than 40 percent, remain in service.

ATTACHMENT

SQN UNIT 2 CYCLE 11 COLD LEG THINNING INDICATIONS

SG	ROW	COL	IND	LOCATION
1	5	1	12	C01+.14
1	30	83	35	C01+.18
1	31	79	27	C01-.31
1	34	19	24	C01+.11
1	34	79	2	C03-.18
1	42	30	21	C01+.20
1	42	62	25	C01-.31
1	43	31	6	C01+.11
1	43	32	8	C01-.02
1	**43**	**60**	**47**	**C01+.00**
1	43	62	12	C01-.22
1	44	36	13	C01+.20
1	44	39	12	C02-.20
1	44	45	19	C02+.04
1	44	58	17	C01-.26
1	44	61	36	C01-.31
1	45	41	31	C02-.27
1	46	44	20	C02-.15
1	46	50	22	C01+.20
1	46	53	15	C01-.18
2	5	93	23	C01-.02
2	5	93	22	C01+.02
2	6	93	22	C01+.14
2	6	94	16	C01-.19
2	6	94	21	C03+.19
2	19	6	19	C01-.04
2	24	8	22	C01-.18
2	24	9	7	C01+.05
2	28	11	29	C01-.20
2	28	12	27	C02-.27
2	33	74	14	C02-.20
2	**33**	**75**	**SVI**	**C02-.10**
2	33	76	33	C02-.25
2	33	78	25	C02-.25
2	34	17	38	C01-.13

SG	ROW	COL	IND	LOCATION
2	34	17	26	C02-.27
2	34	78	22	C02-.22
2	36	18	26	C01+.14
2	36	20	28	C01-.18
2	36	22	15	C01-.22
2	36	76	22	C02-.22
2	**36**	**77**	**SVI**	**C01-.23**
2	38	21	5	C01-.16
2	38	22	26	C03+.00
2	38	24	27	C01+.18
2	38	73	22	C02-.27
2	38	74	6	C02+.11
2	40	24	35	C01-.14
2	40	27	20	C01-.22
2	40	67	23	C02+.11
2	40	68	32	C02-.13
2	40	71	28	C03-.27
2	41	29	26	C01-.02
2	41	33	21	C02-.29
2	41	36	11	C02-.27
2	42	28	17	C01-.21
2	42	29	6	C01+.09
2	42	36	15	C01-.25
2	42	64	25	C01+.11
2	42	66	28	C01+.04
2	43	32	17	C01+.11
2	43	57	23	C02+.18
2	43	63	24	C01-.07
2	**44**	**34**	**SVI**	**C01-.03**
2	44	39	24	C02-.20
2	45	40	22	C01-.14
2	45	43	26	C01-.27
2	45	46	9	C01-.27
2	46	48	15	C01+.09
3	5	1	24	C01+.00
3	7	1	10	C01+.00
3	8	87	20	C02-.05
3	8	87	11	C02+.00
3	9	2	4	C01-.14

SG	ROW	COL	IND	LOCATION
3	10	93	14	C03+.00
3	10	93	34	C01+.00
3	11	92	19	C01-.17
3	25	9	28	C01+.04
3	30	83	14	C01-.28
3	30	83	5	C02-.04
3	31	82	28	C01-.1
3	33	78	20	C01-.19
3	33	79	38	C01-.06
3	34	16	5	C02-.21
3	34	16	20	C01-.17
3	35	17	33	C01-.09
3	36	19	12	C02+.19
3	37	75	26	C01-.06
3	40	26	36	C02-.26
3	**41**	**32**	**46**	**C02-.24**
3	42	28	30	C01-.02
3	**45**	**37**	**48**	**C02-.26**
3	45	52	28	C01-.15
3	45	53	13	C03-.11
4	2	77	11	C07-.92
4	3	25	22	C04-.07
4	7	94	31	C01-.29
4	11	3	12	C01-.27
4	18	13	33	C05-.09
4	19	7	36	C01-.27
4	22	17	23	C04-.11
4	22	17	14	C05-.13
4	22	35	22	C04-.04
4	23	14	7	C05-.04
4	27	11	38	C01-.22
4	30	76	38	C06-.09
4	31	13	15	C01-.27
4	34	76	20	C02+.20
4	34	79	21	C04-.33
4	**35**	**19**	**SVI**	**C02-.08**
4	35	19	30	C02-.36
4	37	19	8	C01-.16
4	39	22	12	C02-.25

SG	ROW	COL	IND	LOCATION
4	39	31	36	C02-.29
4	**39**	**73**	**SVI**	**C02-.12**
4	40	63	27	C02-.09
4	40	71	8	C02-.26
4	41	30	28	C02-.31
4	41	34	18	C02-.27
4	41	58	37	C02-.29
4	41	62	6	C02-.31
4	41	67	36	C01-.29
4	41	69	34	C01-.35
4	42	44	18	C02-.25
4	**43**	**32**	**SVI**	**C01-.12**
4	43	35	1	C02+.09
4	43	54	19	C02-.25
4	44	34	27	C02-.27
4	**44**	**35**	**SVI**	**C01-.13**
4	44	38	32	C02-.20
4	44	47	34	C02-.29
4	45	41	29	C02-.28
4	45	42	12	C02-.33
4	45	45	35	C02-.31
4	45	54	2	C02-.02
4	45	55	13	C01-.33
4	45	57	28	C02-.20
4	45	59	25	C01-.29
4	**46**	**41**	**SVI**	**C02-.10**
4	46	53	10	C02+.16

ENCLOSURE 4

SEQUOYAH NUCLEAR PLANT

REFERENCES

UNIT 2 CYCLE 11 STEAM GENERATOR INSPECTIONS

The following references are associated with Enclosures 1, 2 and 3 of this submittal.

References:

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2. EPRI Report NP-7480-L, Addendum 3, 1999 Database Update, "Steam Generator Outside Diameter Stress Corrosion Cracking at Tube Support Plates - Database for Alternate Repair Criteria," May 1999.

3. EPRI Report NP-7480-L, Addendum 4, 2001 Database Update, "Steam Generator Tubing Outside diameter Stress Corrosion Cracking at Tube Support Plates – Database for Alternate Repair Criteria," March 2001.

4. Letter from G. Srikantiah of EPRI to J. Riley of NEI, "Beaver Valley Data Report", March 28, 2002.

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7. EPRI Report NP-6864-L – Rev. 1, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Expansion Zone PWSCC in Roll Transitions, December 1991.

8. TVA – Sequoyah Nuclear Plant – Unit 2 – "Steam Generator Eddy Current Examination Guideline, Revision 6, November 12, 2003"

9. Westinghouse letter to TVA dated January 5, 2004, Responses to Draft NRC RAIs – Cycle 11 Outage SG Inspection Reports.

10. TVA – Sequoyah Problem Evaluation Report (PER) #04-770190-00 dated January 5, 2004.

11. Westinghouse Report SG-SGDA-01-40, dated December 2002, "Sequoyah Unit 2 Steam Generator Tube Examination"