TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

March 28, 2003

10 CFR 50.90

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING TECHNICAL SPECIFICATION (TS) CHANGE 00-14, "PRESSURE TEMPERATURE LIMITS REPORT (PTLR) AND REQUEST FOR EXEMPTION FROM THE REQUIREMENTS OF 10 CFR 50, APPENDIX G," (TAC NOS. MB6436 AND MB6437)

References: 1. NRC letter to TVA dated February 14, 2003, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Request for Additional Information on Technical Specification (TS) Change No. 00-14, "Pressure Temperature Limits Report (PTLR) and Request for Exemption From the Requirements of 10 CFR 50, Appendix G," (TAC Nos. MB6436 and MB6437)

2. TVA letter to NRC dated December 19, 2002, Sequoyah Nuclear Plant (SQN) - Westinghouse Electric Company Topical Report (WCAP-15984) for Technical Specification Change No. 00-14, "Pressure Temperature Limits Report (PTLR) and Request for Exemption from the Requirements of 10 CFR 50, Appendix G"

The enclosure of this letter provides additional information requested by the letter in Reference 1 to support NRC review of SQN TS Change 00-14.

By the Reference 2 letter, TVA submitted WCAP-15984 for SQN that provided the basis for the exclusion of the reactor vessel head flange region as part of the development of SQN's

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heatup and cooldown limits. TVA anticipates providing a revision to WCAP-15984 that will supersede the information provided by Reference 2. Additionally, revised PTLR's and supporting topical reports will be provided.

As coordinated with the NRC staff, the response date contained in Reference 1 was extended to March 28, 2003, to allow time for development of technical information.

This letter is being sent in accordance with NRC RIS 2001-05. 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,

Pedro Salas
Licensing and Industry Affairs Manager

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 28 day of March, 2003.

Enclosure
cc (Enclosure):

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ENCLOSURE 1

SEQUOYAH NUCLEAR PLANT (SQN)
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)

TECHNICAL SPECIFICATION (TS) CHANGE NO. 00-14,
DOCKET NOS. 50-327 AND 50-328

NRC Question 1

Regarding the discussion on page 4-1, "These results [the stresses for boltup and steady-state operation given in Table 4-1] were taken from a finite element analysis of the heatup/cooldown process, and the boltup was determined to be the most limiting time step for the entire heatup/cooldown transient."

Provide a summary which identifies how the finite element analysis (FEA) was performed, including important analysis variables (e.g., mesh size/element used, convergence criteria, thermal transient time step magnitude, boundary conditions, etc.). The level of detail provided should be such that the U.S. Nuclear Regulatory Commission (NRC) staff will have reasonable assurance regarding the acceptability of the licensee's FEA process and input variables. Provide stress analysis results consistent with the level of detail provided in Table 4-1) and through wall temperature distributions at twenty evenly distributed points along the most limiting heatup/cooldown transient which was analyzed.

TVA Response to NRC Question 1

1.0 Stress Analysis:

The stress analysis was performed using an axisymmetric model shown in cross section in Figure 1-1, where the cross section of interest is highlighted as Section 3.

There are 54 studs that join the reactor vessel head to the closure flange. These studs are equally spaced around the vessel. Two-dimensional axisymmetric elements were used to model the closure flange region of the reactor vessel. There are a total of 996 elements and 1139 nodes in the model.

The bulk of the model is comprised of isotropic elements. Constant strain elements were used for all the orthotropic elements, as well as for any three node isotropic elements.

Four node isoparametric elements were used for all the four node isotropic elements, which comprise the bulk of the model.

To model the nuts, bolts and the flange material between the bolt holes, orthotropic elements were used. These elements were assigned a very low stiffness value in the hoop direction to account for the absence of any circumferential loads between adjacent members.

The stainless steel clad, which covers the internal surfaces of the vessel, was considered to be non-structural, and was not included as part of the finite element mode. The insulating effect of the clad on model temperatures was included by introducing a modified heat transfer coefficient.

1.1 Mechanical Boundary Conditions:

Physically, the reactor vessel shell will displace laterally, and the crown of the head does not displace laterally. To approximate this behavior, the bottom surface of the model in the shell region and the vertical surface of the model at the vessel crown were both assumed to be resting on rollers. This arrangement of restraint is assumed to correspond to the actual behavior of the vessel, and prevents any rigid body motion of the model.

The initial bolt pre-load tensioning is designed to be so large that the mating flanges of the closure head and shell will never be separated by the contained coolant pressure. Because of this design, only bearing stresses can exist at the interface between the mating flanges of the head and shell. When the contained coolant pressure is zero, these bearing stresses exactly balance the bolt pre-load. As the coolant pressure increases, the flange bearing stresses diminish, since the coolant pressure is now helping the flange bearing stresses in opposing the initial bolt pre-load.

In the absence of contained pressure, the bolt pre-load will rotate the two flanges about a pivot point, so as to reduce the gap. With increasing pressure, the two flanges will rotate in the opposite direction, tending to increase the gap.

The above consideration dictates that for the finite element model, the head flange and the shell flange must be mechanically coupled, so as to carry the mating surface bearing load, as well as to permit free rotation of the flange. The bolt pre-load is designed to be much larger than the contained pressure operating loads. It is, therefore, assumed that during normal operating conditions, the mating surfaces will be pivoted, so as to reduce the gap and that the bearing load will be mostly carried near the pivot point. In keeping with this assumption, the head and shell are mechanically coupled only at one corner node nearest the studs, to allow completely free rotations.

Choosing to couple mechanically at only the pivot location, as discussed above, should tend to increase the bending stresses at all the selected Sections 1 through 5 shown on Figure 1-1. Mechanical coupling at more than one node, or all the nodes on the flange mating surface would have prevented free rotation of the flanges, and reduced the bending stresses.

Based on the above, the coupling scheme is judged to be realistic and slightly conservative.

1.2 Thermal Boundary Conditions:

For thermal analysis, all exterior surfaces of the model were assumed to be perfectly insulated and, therefore, adiabatic. Also, no heat flow was assumed through the external gap between the vessel flange and the head flange.

When the inside surface of the vessel is subjected to thermal transients, the primary mechanism of heat transfer is forced convection. The thermal properties of the metal are computed as linear functions of temperature.

A uniform film coefficient was assumed for the entire inside surface of the vessel, which includes the effect of the clad on the convective heat transfer from the coolant to the vessel wall.

All the nodes on the flange mating surfaces were thermally coupled on the finite element model. It was judged that despite the rotations discussed above, the thermal resistance across the flange mating surfaces will not be significant. This thermal resistance is always ignored in reactor vessel stress reports.

1.3 Bolt Pre-Load:

A bolt pre-load was simulated by applying a 500 degrees Fahrenheit (°F) temperature differential between the flange and the bolts. Again a factor was obtained to relate the stresses, due to the dummy load, to the actual stresses caused by the 116.7 kip/inch pre-load in the reactor vessel.

This factor was of the form:

$$f = \frac{actual\ pre-load\ x\ 2\pi r}{A\ (\sigma\ dummy)\ avg.} = .7059$$

Where r = radius of the bolt circle
A = area of the finite element model simulating the bolts
σ = stress obtained by applying the dummy load

The 116.7 kips/inch pre-load from the bolts is the amount prescribed for the Sequoyah Units 1 and 2 reactor pressure vessel, to prevent the coolant water pressure from separating the flange mating surfaces.

1.4 Stress Results:

The stress analysis was carried out with both temperature and pressure varying with time, and for the heatup and cooldown transient, 15 time steps were analyzed. The heatup/cooldown rate was 100°F per hour. The results for each time step are provided in Table 1-1, showing both hoop and axial (meridional) stresses. The stresses shown in Table 1-1 include thermal and pressure stresses, as well as bolt-up stresses for each time step.

To obtain the meridional stresses, the stresses were rotated from the cylindrical coordinate system used in the model, using standard formulae. The hoop stresses were already in the correct orientation, so no rotation was necessary. As may be clearly seen in the table, the meridional stress was the governing component.

A set of stress intensity factor calculations were carried out to determine the governing time step for the analysis, since the stresses were very similar for several time steps. The Raju and Newman solution for an outside flaw in a cylinder was used, with a postulated surface flaw on the outer surface of the head, with a length six times its depth, oriented perpendicular to the maximum meridional stresses. The governing time step was found to be the end of heatup, or the time step at 344.2 minutes.

The stresses for this case are higher than those reported in WCAP-15984, so the report will be revised to include this case. However, the conclusions of the report are not affected by this finding.

NRC Question 2

Regarding the information provided on page 4-2 on the effect of thermal aging:

a. *Provide the chemical composition (weight percent copper and nickel) of the Sequoyah, Unit 1 and Unit 2, reactor pressure vessel (RPV) closure head region materials.*

b. *Recent work supported by the NRC's Office of Nuclear Regulatory Research has led to the development of new RPV embrittlement models which incorporate terms that have the effect of a "thermal aging" (time-at-temperature) function (original work documented in NUREG/CR-6551, "Improved Embrittlement Correlations for Reactor Pressure Vessel Steel"). The most recent version of the proposed embrittlement model is included below, along with suggested input value definitions:*

$$\text{Shift in } RT_{NDT} = A*f(T_c)*f(P)*f_1(\varphi t)+B*f(Ni)*f(Cu)*f_2(\varphi t)+\text{Bias}$$

where:

$$A = 8.86 \times 10^{-17} \text{ for welds}$$
$$9.30 \times 10^{-17} \text{ for forgings}$$
$$12.7 \times 10^{-17} \text{ for plates}$$

$$f(T_c) = \exp\,(19310/[T_c + 460])$$

$$f(P) = (1 + 110*P)$$

$$f_1(\varphi t) = (\varphi t)^{0.4601}$$

$$B = 230 \text{ for welds}$$
$$132 \text{ for forgings}$$
$$156 \text{ for plates}$$
$$206 \text{ for plates in Combustion Engineering fabricated RPVs}$$

$$f(Ni) = (1 + 2.40 * Ni^{1.250})$$

$$f(Cu) = 0, \text{ if } Cu < 0.072 \text{ wt\%}$$
$$= (Cu - 0.072)^{0.659}$$

$$f_2(\varphi t) = 0.5 + 0.5*\tanh([\log\{\varphi t + 4.579\times10^{12}*t) - 18.265]/0.713)$$

$$\text{Bias} = 0, t < 97{,}000 \text{ hrs}$$
$$= 9.4^\circ F,\ t > 97{,}000 \text{ hrs}$$

and: T_c = *In this application, the temperature of the coolant at the RPV flange*

P = Material phosphorous content, wt%

φt = *Neutron fluence at RPV flange at EOL [end of life]*

(10^{15} n/cm^2 as a nominal value, unless information exists which would suggest that the fluence at the flange could be marginally greater)

Ni = Material nickel content, wt%

Cu = Material copper content, wt%

t = Time of full power operation at end of license conditions in hours (nominally 280000 hrs)

Making the conservative assumption that the embrittlement model equation may be directly applied to the evaluation of thermal aging effects for RPV flange materials (with the effective neutron fluence set to a nominally small value), evaluate what the predicted shift in RT_{ndt} *would be for the Sequoyah, Unit 1 and Unit 2, RPV closure head region materials, and provide the predicted final* RT_{ndt} *values for these materials at the current end of license condition for the units.*

Assess what impact these values would have on the conclusions drawn in WCAP-15984 and the licensee's exemption request.

TVA Response to NRC Question 2.

Effects of Thermal Aging on Ferritic Steels in the Closure Head Region:

This analysis was performed in response to a specific request to use the NRC draft correlation to evaluate long-term thermal aging in reactor head materials. The NRC correlation is based on the analysis of irradiated surveillance capsule data. Although there are time dependent terms in the NRC correlation that account for time at temperature effects, the correlation was never intended to be used in this manner. The calculations have been performed for informational purposes only.

The form of the latest NRC draft correlation is complex and involves three major embrittlement terms. The form of the first two terms is mechanistically based, and the terms represent the hardening contribution from small microstructural defects and clusters created during irradiation. The statistical fitting process gave rise to the third embrittlement term. The detailed form of each of the three terms is shown below:

(1) A stable matrix damage (SMD) term that is based on an assumed understanding of matrix damage mechanisms and parameters:

$$SMD = A \exp[19310/(T_c + 460)] [1 + 110 P] (\phi t)^{0.4601} \quad (1)$$

where,

A is a fitting coefficient that is a function of the material type:

$$A = \begin{Bmatrix} 8.86 \times 10^{-17}, \text{welds} \\ 9.30 \times 10^{-17}, \text{forgings} \\ 12.7 \times 10^{-17}, \text{plates} \end{Bmatrix},$$

T_c is the irradiation temperature as estimated from the cold leg temperature (°F),

P is the measured phosphorus content in wt%, and

ϕt is the fluence ($n \cdot cm^{-2}$ for E > 1 MeV).

(2) A copper rich precipitate (CRP) term that is based on the knowledge of copper-enriched clustering that occurs in RPV steels and the assumed mechanistic understanding of key material and irradiation parameters:

$$CRP = B [1 + 2.40 Ni^{1.250}] F(Cu) \cdot G(\phi t) \quad (2)$$

where,

B is a fitting coefficient that is a function of the material type:

$$B = \begin{Bmatrix} 230, \text{welds} \\ 132, \text{forgings} \\ 206, \text{plates in Combustion Engineering vessels} \\ 156, \text{other plates} \end{Bmatrix},$$

Ni is the measured nickel content in wt%,

F(Cu) is a copper term that is a function of the measured copper content (wt%) and material, relative to a saturation level for high copper content:

$$F(Cu) = \begin{cases} 0, Cu \leq 0.072 \text{ wt\%} \\ (Cu - 0.072)^{0.659}, Cu > 0.072 \text{ wt\%} \end{cases},$$

subject to $Cu_{max} = \begin{cases} 0.25, \text{for welds with Linde 80 or Linde 0091 flux} \\ 0.305, \text{for other welds} \end{cases}$, and

$G(\phi t)$ is a fluence function term that also includes a flux-time at temperature parameter (t_f):

$$G(\phi t) = ½ + ½ \tanh \{[\log (\phi t + 4.579 \times 10^{12}\ t_f) - 18.265] / 0.713\}$$

(3) A bias term that has been included to account for an increased shift when the irradiation time is greater than 97,000 hours:

$$Bias = \begin{cases} 0, t_f < 97{,}000 \text{ h} \\ 9.4, t_f \geq 97{,}000 \text{ h} \end{cases}, \ ^\circ F.$$

The overall predictive shift equation is the sum of the three major terms:

TTS = SMD + CRP + *Bias*.

Evaluations

The SMD term does not contain any time dependent variables other than fluence. Even given the most conservative assumptions (Plate, P = 0.03, T_C =500°F) the SMD contributes less than 1°F to the shift at 1×10^{14} n/cm^2.

The evaluation results depend on the copper content of the materials in the closure head region. A study of the material test certificates for both units reveals copper contents of 0.031, 0.13, and 0.33 weight percent Copper, and the evaluations were done for all three.

There is a time adjustment to the fluence factor, $G(\phi t)$, in the CRP term. Based on the parameters of the equation, 10 years of aging is approximately equivalent to a fast neutron exposure of 4×10^{17}n/cm^2. The contribution of this exposure to the shift will depend on the material form (B term), as well as the Cu and Ni contents. The bias term is added to all materials with exposure times greater than 97,000 hours.

The estimates for the Sequoyah Unit 1 and 2 flange chemistries have been compiled assuming a forging material with a Ni content of 0.8 weight percent and a fluence of 10^{14}n/cm^2. The results of this analysis are summarized in Figure 2-1. Three different possible Cu contents are illustrated. No specific copper contents are available for the Sequoyah units, but the 0.2 Cu level exceeds the level normally expected in RPV Forgings.

From this result, it can be seen that the effect of thermal aging on the head material is small.

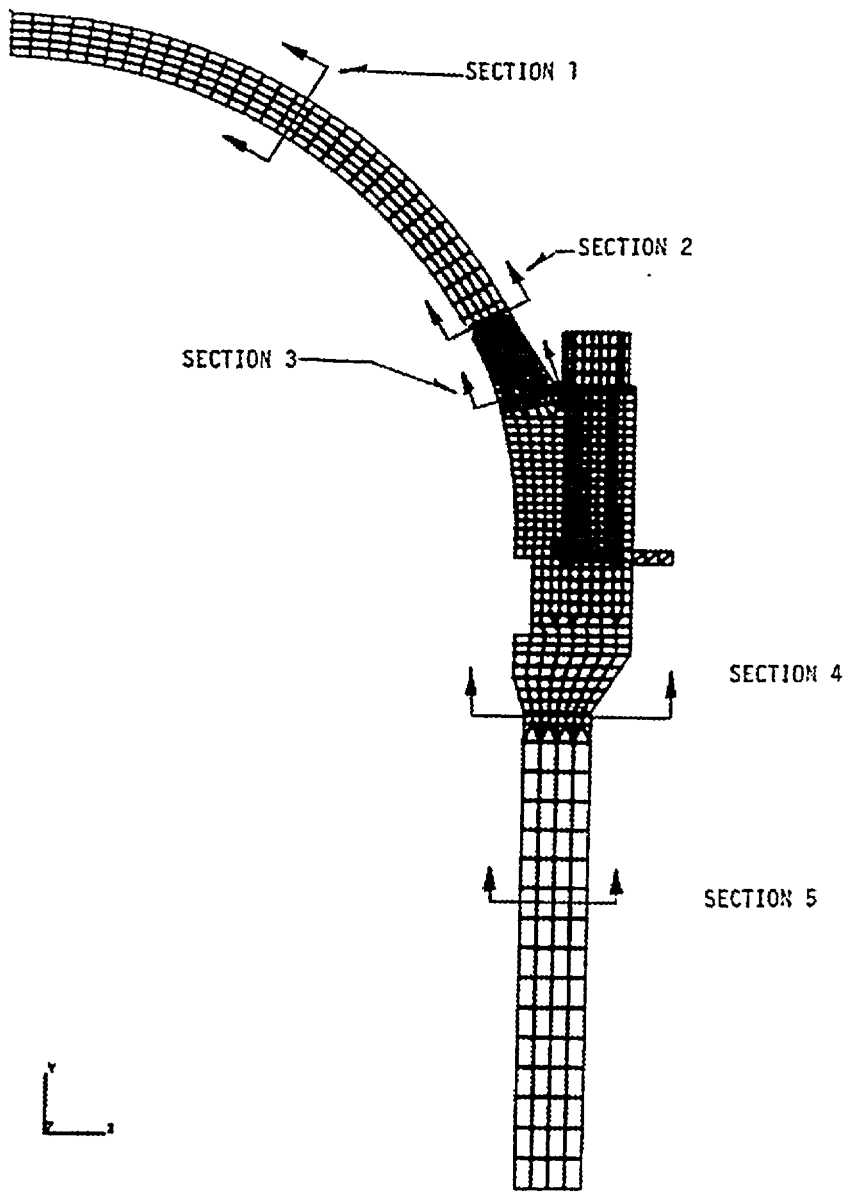

Figure 1-1
Finite Element Model for Closure Head Region,
Sequoyah Units 1 and 2

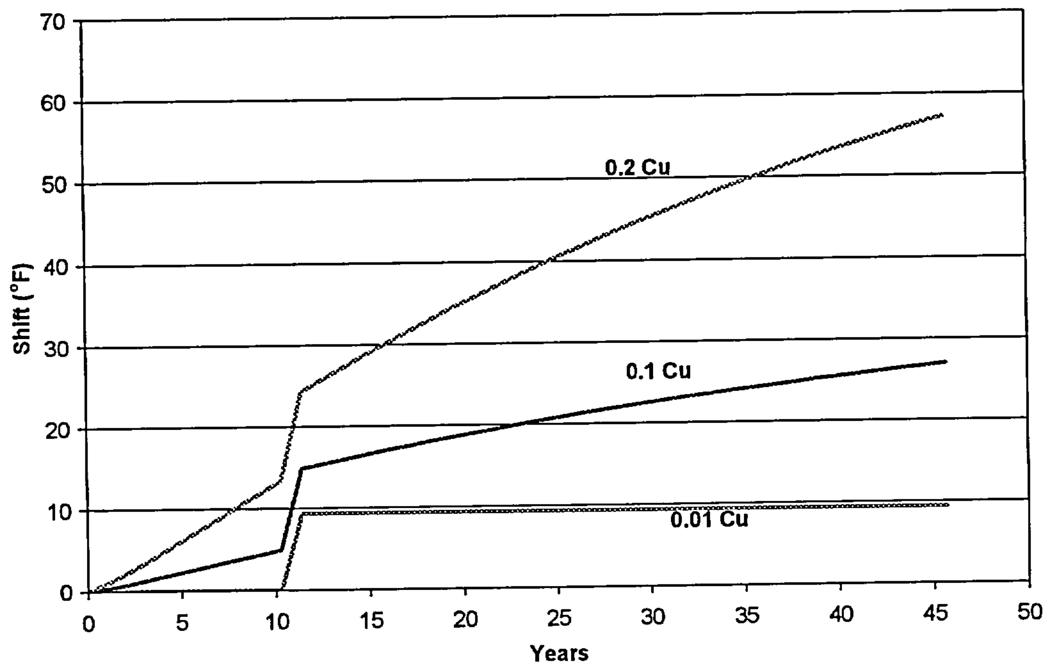

Figure 2-1

NRC Draft Correlation predictions for long time exposures assuming a negligible neutron fluence. Calculation also assumes a forging material with a Ni content of 0.8 weight percent.

Table 1-1:

Stress results for Upper Head to Flange Transition Region

Time (minutes)	Distance through wall (a/t)	Axial Stress with Boltup (psi)	Hoop Stress with Boltup (psi)
82	0.0	-11973	5979.0
(Begin of Heatup)	0.2	-5729.7	3411.8
	0.4	-812.11	887.5
	0.6	-486.45	-1615.2
	0.8	9100.9	-4092.0
	1.0	21133	-6590.5
207.4	0.0	-22542.8	-11774
	0.2	-9090.8	-5611.8
	0.4	-473.08	-2816.2
	0.6	1790.2	-1607.8
	0.8	11730	-1840.0
	1.0	22786	-3514.0
344.2	0.0	-16467	-12530.5
(End of Heatup)	0.2	-789.9	-2572.0
	0.4	8461.7	2641.5
	0.6	10996	5702.0
	0.8	20771	6784.2
	1.0	31477	5746.5
355	0.0	-13064	-8613.5
	0.2	-312.4	-1778.0
	0.4	8014.8	2305.8
	0.6	10323	4958.0
	0.8	20266	5978.2
	1.0	31470	5096.5
375	0.0	-9401.5	-3442
	0.2	912.7	412
	0.4	7776.5	2543.5
	0.6	9329.8	3900.8
	0.8	19226.3	4271.5
	1.0	31216	3410.5

Table 1.1, continued

Time (minutes)	Distance through wall (a/t)	Axial Stress with Boltup (psi)	Hoop Stress with Boltup (psi)
405	0.0	-6491.0	1150.5
	0.2	1958.3	2562.8
	0.4	7661.9	2999.0
	0.6	8552.0	3110.2
	0.8	18364	2742.0
	1.0	30983	1738.0
406	0.0	-180.4	16972
(Begin cooldown)	0.2	4587.0	13742
(Steady State)	0.4	8274.2	10764
	0.6	7749.6	7915.8
	0.8	16762	5220.5
	1.0	28186	2397.5
411	0.0	1000.8	18562
	0.2	3945.6	13355
	0.4	7322.7	9914.2
	0.6	6844.2	6973.2
	0.8	16009	4293.2
	1.0	27676	1548.0
416	0.0	1783.4	19999
	0.2	3639.2	13469
	0.4	6466.9	9261.0
	0.6	5830.7	5952.2
	0.8	15092	3157.0
	1.0	27075	478.5
421	0.0	2209.2	21136
	0.2	3352.9	13642
	0.4	5734.9	8753.5
	0.6	4879.9	4999.2
	0.8	14162	1993.2
	1.0	26427	-665.0

Table 1.1, continued

Time (minutes)	Distance through wall (a/t)	Axial Stress with Boltup (psi)	Hoop Stress with Boltup (psi)
474.4	0.0	-806.0	23866
	0.2	-2216.0	11823
	0.4	-1357.3	3481.5
	0.6	-3115.1	-2978.8
	0.8	6245.0	-7695.8
	1.0	19929	-10692
668.2	0.0	-157.6	29057
(End Cooldown)	0.2	-1344.7	17292
	0.4	-473.0	9039.5
	0.6	-2633.4	1932.5
	0.8	5915.3	-3684.2
	1.0	17437	-7982.5
678	0.0	-2528.0	26011
	0.2	-1829.6	16634
	0.4	-207.3	9099.5
	0.6	-2155.8	2472.2
	0.8	6367.1	-2961.2
	1.0	17520	-7309.0
698	0.0	-5294.9	21748
	0.2	-2860.4	14618
	0.4	-106.3	8676.2
	0.6	-1366.5	3186.8
	0.8	7224.3	-1607.0
	1.0	17859	-5859.5
728	0.0	-7246.5	18350
	0.2	-3678.8	12759
	0.4	-150.6	7992.5
	0.6	-885.3	3463.2
	0.8	7882.4	-682.5
	1.0	18303	-4679.0

NRC Question 3

Regarding the discussion on page 5-1 on the basis for the reference flaw size:

Provide information that explains what RPV head flange region inservice inspections (when the inspections were conducted, the extent of coverage achieved, ultrasonic transducers used, etc.) have been conducted at Sequoyah, Units 1 and 2, relative to the discussion in WCAP-15984 regarding the quality of inspections cited to support the assumed reference flaw size. More specifically, provide an evaluation that demonstrates how the inspections conducted at Sequoyah, Units 1 and 2, support the assumption of a 0.1T flaw size in the flange evaluation.

TVA Response to NRC Question 3

The Sequoyah Unit 1 and 2 reactor vessel closure head to flange welds were examined during the present 10 year in-service inspection interval to the requirements of the 1989 Edition of American Society of Mechanical Engineers (ASME) Section XI. The inspections were conducted as follows.

Unit 1:

> Approximately 50% of the weld length was examined on September 19, 1998. (2^{nd} interval, 1^{st} period).
>
> The remainder of the weld was examined on March 1, 2000. (2^{nd} interval, 2^{nd} period).

Unit 2:

> Approximately 50% of the weld length was examined on April 25-26, 1999. (2^{nd} interval, 1^{st} period).
>
> The remainder of the weld was examined on October 31, 2000. (2^{nd} interval, 2^{nd} period).

As discussed above, the inspection procedure was prepared to meet the requirements of the 1989 Edition of ASME Section XI and included requirements for both a 100% volumetric and 100% surface examination.

The volumetric inspection was performed using an ultrasonic examination as required by the Code. The ultrasonic examination was performed from three directions (transverse from head side of the weld and two parallel directions over the weld). An average of 80% coverage for Unit 1 and 67% coverage for Unit 2 was achieved. This examination was demonstrated to the satisfaction of the TVA Authorized Nuclear Inspector (Hartford Steam Boiler). A 100% volumetric weld inspection was not performed because the geometric configuration of the flange-to-head weld precluded a complete examination. This issue was evaluated by NRC as part of TVA Relief Requests 1-ISI-2 (Part 1) and 2-ISI-2 (Part 1) and was found to be acceptable by the Safety Evaluation Report enclosed in the NRC letter from R. W. Hernan to O. J. Zeringue dated April 27, 1998.

Digitized ultrasonic pulse echo instruments with one inch diameter, 2.25 MHz transducers with plastic wedges were used for the 45 and 60-degree angle beam examinations. A one inch diameter, 2.25 MHz 0 degree transducer was used for the straight beam examination. A basic 7-inch calibration block (cladded on one side) was used to calibrate the equipment. As required, 5/16 inch diameter side drilled holes and 2% notches (depth of approximately 0.140 inch) were used for the ultrasonic calibrations. The wave forms and beam spread plots were performed prior to the examinations. Resolution verifications were also performed. During the examination, scanning sensitivity was performed at a level at least +6 dB above the reference level. In addition, to ensure sound beam penetration, the gain control was adjusted to display an opposite surface noise level of approximately 5% full screen height.

The surface examination of the weld and flex area was performed using magnetic particle inspection techniques. Full coverage (100%) of the weld and flex area was achieved. This examination was demonstrated to the satisfaction of the TVA Authorized Nuclear Inspector (Hartford Steam Boiler).

The inspection procedure required flaw sizes to be established using the standards of IWA-3000. Evaluation of any flaws was to be performed to Table IWB-3510 (as required under Category B-A of Table IWB-2500). No indications were found using either of the two inspection techniques described above.

The thickness of the closure head to flange weld for Sequoyah Units 1 and 2 is nominally 7.24 inches. Based on the magnetic particle examination and ultrasonic recording and evaluation criteria required by the inspection procedure, there is a high degree of confidence that a 0.1T (0.73 inch) flaw size would have been detected during the in-service inspection.

<u>NRC Question 4</u>

Regarding Item 2.1 .2.a. in each unit's PTLR, it is stated that a maximum heatup rate of 100°F in any one-hour period is permitted. However, a heatup limit curve (Fig. 2-1 in either PTLR) is only *given for a rate of 60^0F per hour. Either modify Item 2.1.2.a in each proposed PTLR or explain why Item 2.1.2.a in each unit's PTLR should not be changed to 60^0F in any one-hour period.*

TVA Response to Question 4

For consistency with the text in Item 2.1.2.a, Figure 2-1 in each PTLR will be revised to reflect a maximum heat up rate of 100°F/hr rather than 60°F/hr. Final revisions to the Sequoyah PTLRs are in progress and will be submitted to NRC by separate correspondence.

NRC Question 5

Regarding Section 4.0 in each unit's PTLR, why is American Society for Testing and Materials (ASTM) Standard E208 (on nil-ductility reference temperature testing) noted? It would seem that, in terms of an applicable ASTM Standard which might be of interest under the subject of "Reactor Vessel Material Surveillance Program" that ASTM Standard E23 (on notched bar impact testing), if anything, would be a more suitable reference. Either modify Section 4.0 in each unit's PTLR or explain why the current reference to ASTM E208 is considered to be more appropriate.

TVA Response to NRC Question 5

We concur that a reference to ASTM Standard E23 is more accurate than the present reference ASTM Standard E208. Section 4.0 and the appropriate reference in Section 6.0 of each PTLR will be revised accordingly. Final revisions to the Sequoyah PTLRs are in progress and will be submitted to NRC by separate correspondence.

NRC Question 6

In Tables 5-1, 5-2, and 5-3 of each unit's PTLR, it would be more clear if the heat numbers for each surveillance material and RPV forgings and welds were provided in each table. The heat-to-heat association of surveillance material with RPV materials is a critical component in the Regulatory Guide 1.99, Revision 2 (RG 1.99, Rev. 2) evaluation process.

TVA Response to NRC Question 6

Heat numbers for each surveillance material, reactor pressure vessel forging and welds will be added to Tables 5-1, 5-2 and 5-3 of the PTLRs. Final revisions to the Sequoyah PTLRs are in progress and will be submitted to NRC by separate correspondence.

NRC Question 7a and 7b

Regarding Tables 5-5, 5-6, and 5-7 of each unit's PTLR;

a. *Why are margin terms in the table for the use of Position 1.1 and Position 2.1 shown to be equivalent? Per RG 1.99, Rev. 2, when credible surveillance data is used, the "6? term" in the margin calculation may be halved. It is not clear how the results shown in the tables for "Position 2.1" are consistent with PTLR methodology cited in WCAP-14040-A, Rev. 2. If the licensee's evaluation is intending to reference additional NRC staff guidance into the evaluation of each unit's surveillance data (in which case the evaluation given as "Position 2.1" for each material may not actually follow the specific outline of Position 2.1 in RG 1.99, Rev. 2), this additional staff guidance should be clearly referenced in each unit's PTLR methodology.*

b. *Based on the proposed Sequoyah, Unit 1 and Unit 2, PTLR methodology, the results from the evaluation of whether the Sequoyah, Unit 1 and Unit 2, surveillance data is, or is not, credible should be clearly stated in the PTLR, although the actual calculations which support the credibility evaluation may be referenced from elsewhere. Based on this determination, there should also be an indication given in Tables 5-5, 5-6, and 5-7 of which position (1.1 or 2.1) is considered by the licensee to be the licensing basis calculation for each material.*

TVA Response to NRC Question 7a and 7b

Tables 5-5, 5-6, and 5-7 of the PTLRs will be revised to clearly state the basis for the margin calculation. Final revisions to the Sequoyah PTLRs are in progress and will be submitted to NRC by separate correspondence.

NRC Question 8

It appears that throughout each unit's PTLR, conservative fluence values have been used (i.e., a single, peak fluence location was determined for the entire vessel then that value was used for the evaluation of all RPV materials). Confirm whether or not this understanding is correct. If so, at a minimum, a footnote should be added to Tables 5-5, 5-6, and 5-8, which explains that the neutron fluence values cited in the PTLR are not the actual, calculated peak values for each forging or weld.

TVA Response to Question 8

A single peak fluence location was established for the reactor vessel beltline region and the fluence for that location was used in the evaluation of all reactor pressure vessel materials. The various locations and the projected peak fluence values for those locations are shown in Table 5-4 of each PTLR.

Tables 5-5, 5-6 and 5-8 in each PTLR are for the 32 effective full power years (EFPY) projected fluence case. The 32 EFPY peak fluence value in Table 5-4 was used for evaluation of all reactor pressure materials. These tables will be revised to include footnotes as discussed in the RAI. Final revisions to the Sequoyah pressure temperature limits report (PTLR) are in progress and will be submitted to NRC by separate correspondence.

NRC Question 9

Regarding Table 2-2 in each unit's PTLR, provide the 1/4 T K_{IT} and 1/4 T metal temperature for each data point listed for the 100 ^{0}F per hour cooldown transient curve.

TVA Response to NRC Question 9

Tables 1 and 2 provide the respective Unit 1 and Unit 2 ¼ T K_{IT} and ¼ T metal temperature for the 100°F per hour cooldown transient curve.

Table 1 - Sequoyah Unit 1
K_{IT} Values for 100°F/hr Cooldown Curve (32 EFPY)

Water Temp (°F)	1/4T Wall Temp. (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
220	246.07[a]	16.3635[a]
215	240.99	16.2981
210	235.90	16.2331
205	230.82	16.1673
200	225.74	16.1018
195	220.65	16.0357
190	215.57	15.9700
185	210.49	15.9038
180	205.40	15.8379
175	200.32	15.7715
170	195.23	15.7056
165	190.15	15.6393
160	185.07	15.5734
155	179.98	15.5070
150	174.90	15.4412
145	169.81	15.3750
140	164.73	15.3092
135	159.64	15.2432
130	154.56	15.1776
125	149.48	15.1117
120	144.39	15.0463
115	139.31	14.9806
110	134.22	14.9154
105	129.14	14.8500
100	124.06	14.7850
95	118.97	14.7198
90	113.89	14.6551
85	108.81	14.5902
80	103.72	14.5258
75	98.64	14.4611
70	93.56	14.3969
65	88.48	14.3325
60	83.39	14.2678
55	78.32	14.2020
50	73.24	14.1362

Note:

(a) Temperatures at 220°F and higher are limited by a Lower Rate and/or Steady State

Table 2 - Sequoyah Unit 2
K_{IT} Values for 100°F/hr Cooldown Curve (32 EFPY)

Water Temp (°F) (a)	1/4T Wall Temp. (°F) (a)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
145	(b)	(b)
140	164.73	15.3092
135	159.64	15.2432
130	154.56	15.1776
125	149.48	15.1117
120	144.39	15.0463
115	139.31	14.9806
110	134.22	14.9154
105	129.14	14.8500
100	124.06	14.7850
95	118.97	14.7198
90	113.89	14.6551
85	108.81	14.5902
80	103.72	14.5258
75	98.64	14.4611
70	93.56	14.3969
65	88.48	14.3325
60	83.39	14.2678
55	78.32	14.2020
50	73.24	14.1362

Notes:
(a) Does NOT include any Temperature Instrumentation Margins
(b) Temperatures at 145°F and higher are limited by a Lower Rate and/or Steady State

NRC Question 10

Regarding Table 2-1 in each unit's PTLR, provide the 1/4 T metal temperature, 3/4 T K_{IT} and 3/4 T metal temperature for each data point listed for the 60°F per hour heatup transient curve.

TVA Response to NRC Question 10

Tables 3 and 4 provide the respective Unit 1 and Unit 2, 1/4 T K_{IT} and 1/4 T metal temperature for the 100°F per hour heatup transient curve. The 100°F values rather than 60°F values were provided to be consistent with the response provided to Question No. 4.

Table 3 - Sequoyah Unit 1
K_{IT} Values for 100°F/hr Heatup Curve (32 EFPY)

Water Temp (°F) (a)	1/4T Wall Temp. (°F) (a)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	3/4T Wall Temp. (°F) (a)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
Minimum Pressure for Heatup Curves Occurs at T = 150°F (b)				
150	130.10	-11.8741	109.21	8.9173
155	134.72	-12.1429	113.34	9.1310
160	139.38	-12.3911	117.56	9.3251
165	144.06	-12.6135	121.85	9.5015
170	148.78	-12.8203	126.21	9.6628
175	153.52	-13.0068	130.62	9.8103
180	158.28	-13.1814	135.08	9.9461
185	163.06	-13.3400	139.59	10.0712
190	167.86	-13.4897	144.13	10.1871
195	172.67	-13.6266	148.71	10.2948
200	177.49	-13.7569	153.32	10.3954
205	182.33	-13.8770	157.96	10.4896
210	187.17	-13.9921	162.62	10.5782
215	192.03	-14.0992	167.30	10.6618
220	196.89	-14.2026	171.99	10.7412
225	201.76	-14.2994	176.71	10.8166
230	206.63	-14.3937	181.44	10.8887
235	211.51	-14.4826	186.18	10.9577
240	216.40	-14.5698	190.93	11.0242
245	221.29	-14.6526	195.69	11.0882
250	226.18	-14.7342	200.46	11.1503
255	231.08	-14.8122	205.24	11.2105
260	235.98	-14.8895	210.02	11.2692
265	240.88	-14.9638	214.81	11.3264
270	245.78	-15.0377	219.61	11.3824
275	250.69	-15.1092	224.41	11.4373
280	255.60	-15.1806	229.21	11.4913
285	260.50	-15.2498	234.02	11.5444
290	265.42	-15.3192	238.83	11.5968
295	270.33	-15.3868	243.64	11.6485
300	275.24	-15.4546	248.46	11.6997
305	280.15	-15.5209	253.27	11.7504
310	285.07	-15.5876	258.09	11.8007
315	289.98	-15.6529	262.91	11.8505
320	294.90	-15.7187	267.74	11.9001
325	299.81	-15.7833	272.56	11.9494
330	304.73	-15.8484	277.39	11.9984
Heatup limited by 3/4T from T = 150°F to T = 330°F				

Notes:

(a) Does Not include any Temperature Instrumentation Margins

(b) Minimum Pressure occurs at 150°F and thus carried down to 50°F

- Note that the Vessel Radius to the ¼T and ¾T Locations are as follows:
  1/4T Radius = 88.771 inches 3/4T Radius = 92.996 inches

Table 4 - Sequoyah Unit 2
K_{IT} Values for 100°F/hr Heatup Curve (32 EFPY)

Water Temp (°F) [(a)]	1/4T Wall Temp. (°F) [(a)]	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	3/4T Wall Temp. (°F) [(a)]	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
Heatup limited by Steady State up to T = 85°F, Remaining Curve Limited by 3/4T Location				
90	80.03	-4.8425	72.23	3.3275
95	83.61	-5.8495	73.85	4.1378
100	87.29	-6.7697	75.86	4.8626
105	91.17	-7.5681	78.20	5.5042
110	95.14	-8.2935	80.85	6.0763
115	99.24	-8.9270	83.75	6.5842
120	103.43	-9.5025	86.89	7.0378
125	107.72	-10.0077	90.22	7.4419
130	112.07	-10.4676	93.74	7.8040
135	116.50	-10.8733	97.42	8.1278
140	120.98	-11.2440	101.23	8.4191
145	125.52	-11.5725	105.16	8.6808
150	130.10	-11.8741	109.21	8.9173
155	134.72	-12.1429	113.34	9.1310
160	139.38	-12.3911	117.56	9.3251
165	144.06	-12.6135	121.85	9.5015
170	148.78	-12.8203	126.21	9.6628
175	153.52	-13.0068	130.62	9.8103
180	158.28	-13.1814	135.08	9.9461
185	163.06	-13.3400	139.59	10.0712
190	167.86	-13.4897	144.13	10.1871
195	172.67	-13.6266	148.71	10.2948
200	177.49	-13.7569	153.32	10.3954
205	182.33	-13.8770	157.96	10.4896
210	187.17	-13.9921	162.62	10.5782
215	192.03	-14.0992	167.30	10.6618
220	196.89	-14.2026	171.99	10.7412
225	201.76	-14.2994	176.71	10.8166
230	206.63	-14.3937	181.44	10.8887
235	211.51	-14.4826	186.18	10.9577
240	216.40	-14.5698	190.93	11.0242
245	221.29	-14.6526	195.69	11.0882
250	226.18	-14.7342	200.46	11.1503
255	231.08	-14.8122	205.24	11.2105

Note:

(a) Does <u>NOT</u> include any Temperature Instrumentation Margins.

- Note that the Vessel Radius to the ¼T and ¾T Locations are as follows:

1/4T Radius = 88.771 inches

3/4T Radius = 92.996 inches

NRC Question 11

Has the information that is currently in Sequoyah, Unit 1 and Unit 2, Final Safety Analysis Reports been reconciled with information in the PTLR methodology documentation (e.g., information on fluence calculation methodology)? If a reconciliation of information in the PTLR methodology has been completed, please state so. If not, please summarize your process for ensuring that such a reconciliation will be completed in a timely manner relative to the issuance of the PTLR.

TVA Response to NRC Question 11

The fluence calculation methodology currently described in the Final Safety Analysis Report (FSAR) will be reconciled with the fluence calculation methodology in the PTLR methodology documentation as part of the site implementation of the proposed TS change. The TS amendment process ensures documents required for TS or license amendment are complete and ready for simultaneous implementation with the approved amendment. As such, the FSAR will be revised to reflect the PTLR methodology documentation in accordance with the existing plant design change process required to authorize use of the revised reactor coolant system heatup and cooldown limits and low temperature overpressure protection setpoints for amendment implementation following the approval of the license amendment request.

NRC Question 12

Consistent with the guidance in Generic Letter (GL) 96-03, it appears that all Sequoyah, Unit 1 and Unit 2, surveillance capsule reports should be clearly referenced in each unit's respective PTLR (see the Table in GL 96-03, Item 2, Column 3). The licensee's current submittal does not include all of these references. The licensee should either add the appropriate references or explain why they are not necessary.

TVA Response to NRC Question 12

Section 6.0 in the PTLRs will be revised to reference all the Sequoyah surveillance capsule reports. Final revisions to the Sequoyah PTLRs are in progress and will be submitted to NRC by separate correspondence.

NRC Question 13

Consistent with GL 96-03 Table, Item 6, Column 3, specific minimum temperature requirements should be listed on the pressure temperature limit figures in the PTLRs. For clarity, the licensee should consider whether the numeric value for boltup temperature (50°F) should be added to the figure in each PTLR.

TVA Response to NRC Question 13

The numeric value for the minimum boltup temperature will be added to the heatup and cooldown curves in the PTLRs. Final revisions to the Sequoyah PTLRs are in progress and will be submitted to NRC by separate correspondence