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

April 16, 2007

10 CFR 50.55(a)

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

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

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2 - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) FOR RISK-INFORMED INSERVICE INSPECTION (RI-ISI) PROGRAM UPDATE

Reference: NRC letter to TVA dated March 29, 2007, "Sequoyah Nuclear Plant, Units 1 and 2 - Request for Additional Information Regarding Risk-Informed Inservice Inspection Program Relief Requests (TAC Nos. MD1452, MD1453, MD1454 and MD 1455)"

The purpose of this letter is to provide TVA's response to the reference letter. TVA's response to the RAI supports NRC review of the SQN Risk-Informed Inservice Inspection (RI-ISI) program. The RI-ISI program is applicable to SQN's third 10-year inspection interval.

There are no commitments contained in this submittal.

Please direct questions concerning this issue to me (423) 843-7170.

Sincerely,

Glenn W. Morris

Glenn W. Morris
Manager, Site Licensing and
Industry Affairs

Enclosure
cc: See page 2

A047

Printed on recycled paper

cc (Enclosures):

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ENCLOSURE

SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

The following provides TVA's response to NRC's request for additional information letter dated March 29, 2007.

NRC Question 1

Noting that the Sequoyah Nuclear Plant (SQN) Probabilistic Risk Assessment (PRA) Revision 1 model was used in support of the original risk-informed inservice inspection program (RI-ISI) relief in March 2001; the SQN PRA Revision 2 model underwent the Westinghouse Owner's Group (WOG) PRA Peer Review Certification process; and the SQN PRA Revision 3 is being used for this recent RI-ISI relief request. As stated in the Regulatory Guide 1.178: "A description of the staff and industry reviews performed on the PRA. Limitations, weakness, or improvements identified by the reviewers that could change the results of the PRA should be discussed. The resolution of the review comments, or an explanation of the insensitivity of the analysis should be provided." However, in the current relief request, there is no discussion of PRA updates. Hence, to establish confidence that the quality of the SQN PRA Revision 3 is sufficient to support this RI-ISI relief, please answer the following questions and/or provide the information requested below.

a. **Confirm the dates of SQN Revision 3 PRA models for both Units 1 and 2.**

b. **Were all of the Level A and B Facts & Observations (F&Os) from the WOG PRA Peer Review Certification resolved and/or incorporated into the PRA model(s) used for this application (i.e., Units 1 and 2 SQN PRA Revision 3)? If not, identify and state why the unincorporated F&Os are not expected to have an impact on the RI-ISI program.**

c. **Provide the baseline core damage frequencies and large early release frequencies (for both Units 1 and 2) from SQN PRA Revision 3.**

TVA Response

a. The SQN Revision 3 PRA Model used in the development of the current RI-ISI relief request is dated August, 2004 (see page 2 of the cover letter for TVA's April 21, 2006 RI-ISI relief request).

b. In the original SQN RI-ISI relief request, the SQN Revision 1 PSA model was used and the NRC staff concluded that it was of sufficient quality to be used in the RI-ISI application (see Section 3.3 of the Safety Evaluation Report (SER) for the original SQN RI-ISI relief). None of the F&Os from the WOG PRA Peer Review Certification were resolved and/or incorporated into the SQN Revision 1 PSA Model. All of the level A F&Os and the more significant level B F&Os from the WOG PRA Peer Review Certification have been resolved and/or incorporated into the SQN Revision 3 PSA Model. Therefore, the SQN Revision 3 PSA Model is of higher quality than the SQN Revision 1 Model used for the original RI-ISI relief request.

In addition, the results of the PSA consequence analysis for this RI-ISI relief request demonstrate that the relative importance of pipe segment failures behave as expected:

- Pipe segment failures that cause the loss of a train of a risk-significant system result in a larger increase in core damage frequency (CDF) and Large early release frequency (LERF) compared to other systems. The risk-significant systems for SQN are the same as for all pressurized water reactors (PWRs) (i.e., see those systems in the scope of Mitigating System Performance Index [MSPI]).

- Pipe segment failures that cause the loss of both trains of a MSPI system or cause the loss of multiple MSPI systems, result in some of the largest increases in CDF and LERF.

- Pipe segments whose failure causes degradation in the ability to isolate a large containment penetration result in a relatively large increase in LERF compared to those pipe segments whose failure does not affect containment performance.

Based on the above, the SQN Revision 3 PSA Model is of sufficient quality to be used in support of the current relief request.

c. The SQN Revision 3 PRA Model used in the development of the current RI-ISI relief request has a baseline CDF of 1.31E-05 and a baseline LERF of 2.62E-07.

NRC Question 2

Please update the following table, which was completed and provided by the Tennessee Valley Authority in a response (dated August 31, 2001) to the U. S. Nuclear Regulatory Commission request for additional information (RAI) on the original RI-IS1 program (dated July 13, 2001), with the SQN Revision 3 PRA results.

System	Number of Segments with Any RRW ≥ 1.005	Number of Segments with Any RRW Between 1.005 and 1.001	Number of Segments with Any RRW ≥ 1.005 but Placed in LSS	Number of Segments with All RRW < 1.005 but Placed in HSS

TVA Response

Tables 2-1 and 2-2 below provide the requested information for the analyses performed for the third ten-year ISI program.

Table 2-1 SQN Unit 1				
System ID*	Number of Segments with Any RRW > 1.005	Number of Segments with Any RRW Between 1.005 and 1.001	Number of Segments with Any RRW > 1.005 but Placed in LSS	Number of Segments with All RRW < 1.005 but Placed in HSS
AF	6	6	2	0
BD	6	10	0	0
CH	10	4	0	0
CI	0	0	0	0
CS	4	0	2	1
FW	0	8	0	0
MS	4	1	0	0
RC	5	40	0	2
RH	0	11	0	5
SI	20	24	0	11
SQ	0	5	0	0
Total	55	109	4	19

Table 2-2 SQN Unit 2				
System ID*	Number of Segments with Any RRW > 1.005	Number of Segments with Any RRW Between 1.005 and 1.001	Number of Segments with Any RRW > 1.005 but Placed in LSS	Number of Segments with All RRW < 1.005 but Placed in HSS
AF	6	6	2	0
BD	0	16	0	6
CH	10	4	0	0
CI	0	0	0	0
CS	4	2	2	1
FW	0	8	0	0
MS	4	1	0	0
RC	5	40	0	1
RH	0	11	0	5
SI	20	23	0	9
SQ	0	5	0	0
Total	49	116	4	22

*The system IDs are defined below:

AF = Auxiliary Feedwater
BD = Steam Generator Blowdown
CH = Charging
CI = Containment Isolation
CS = Containment Spray
FW = Feedwater
MS = Main Steam
RC = Reactor Coolant
RH = Residual Heat Removal
SI = Safety Injection
SQ = Sampling and Water Quality

NRC Question 3

For the 72 pipe segments of SQN Units 1 and 2 that were moved from being in the category of high safety significant (HSS) to low safety significant (LSS) and the 31 LSS segments of Units 1 and 2 that were increased to HSS category:

a. Please identify these pipe segments that moved from HSS to LSS and provide an explanation why each segment's safety significance changed. Please summarize the changes that caused other pipe segments to move from LSS to HSS.

TVA Response

Tables 3-1 and 3-2 below provide a qualitative summary of information for segments changing from HSS in the original RI-ISI program to LSS for the update. For each segment in the tables, the three primary categories of data input to the risk evaluation are presented with the change described in each category. The comment columns in Tables 3-1 and 3-2 provide the primary reason for the changes that resulted in the LSS categorization. Note that the PRA has been revised more than once since the original submittal, so previous changes have also affected the results. Tables 3-3 and 3-4 provide a qualitative summary of information for segments changing from LSS in the original RI-ISI program to HSS for the update. The comment columns for Tables 3-3 and 3-4 provide the primary reasons for the categorization change when the information was readily available. Note that the system IDs are defined in the above response to NRC Question 2.

Table 3-1 SQN Unit 1 Segments that Changed From HSS to LSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
AF	None				
BD	BD-005	CDF and LERF values decreased	No change	No change	Improvement in the reliability and availability of equipment used to mitigate secondary side breaks (AFW) resulted in a decrease in CDF and LERF.
BD	BD-006	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-007	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-008	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-010	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-011	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-018	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-019	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
CH	CH-018	CDF and LERF values decreased	No change	No change	Changes to PRA results after the initial expert panel meetings resulted in RRWs that supported a LSS ranking. Due to timing considerations, the expert panel chose to leave this segment as HSS for the original submittal. Revised inputs support LSS ranking.
CH	CH-019	CDF and LERF values decreased	No change	No change	See Comment for CH-018.
CH	CH-020	CDF and LERF values decreased	No change	No change	See Comment for CH-018.
CI	None				
CS	None				

Table 3-1 SQN Unit 1 Segments that Changed From HSS to LSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
FW	FW-005	LERF values decreased	No change	No change	Final RRWs for original submittal were less than 1.005, but the segment was ranked HSS. Segment was originally HSS because of LERF results. LERF results decreased primarily because of an updated operator action (backup to actuation signals) in the model. The HRA calculator was used in the first major update to the HRA since 1992.
FW	FW-006	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-007	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-008	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-009	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-010	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-011	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-012	LERF values decreased	No change	No change	See Comment for FW-005.
MS	None				
RC	RC-017	CDF and LERF values decreased	No change	No change	Final RRWs for original submittal were less than 1.005, but the segment was ranked HSS. Primary contributor to the change is the recalculation of the operator action failure probability for aligning high pressure recirculation. The HRA calculator was used in the first major update to the HRA since 1992.
RC	RC-018	CDF and LERF values decreased	No change	No change	See Comment for RC-017.
RC	RC-020	CDF and LERF values decreased	No change	No change	See Comment for RC-017.
RC	RC-025	CDF and LERF values decreased	No change	No change	See Comment for RC-017.

Table 3-1 SQN Unit 1 Segments that Changed From HSS to LSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
RC	RC-027	CDF and LERF values decreased	No change	No change	See Comment for RC-017.
RC	RC-028	CDF and LERF values decreased	No change	No change	See Comment for RC-017.
RC	RC-029	CDF and LERF values decreased	Decreased	No change	See Comment for RC-017. In addition, the failure probability decreased because the ISI failure probability was used. The segment was added to the augmented inspection program for Alloy 600.
RC	RC-047	CDF and LERF values decreased	No significant change	No change	See Comment for RC-017.
RC	RC-048	CDF and LERF values decreased	No significant change	No change	See Comment for RC-017.
RC	RC-049	CDF and LERF values decreased	No significant change	No change	See Comment for RC-017.
RC	RC-051	CDF and LERF values decreased	No change	No change	See Comment for RC-017.
RH	None				
SI	SI-021B	CDF and LERF values decreased	Increased	No change	Changes to PRA results after the initial expert panel meetings resulted in RRWs that supported a LSS ranking. Due to timing considerations, the expert panel chose to leave this segment as HSS for the original submittal. Revised inputs support LSS ranking.
SI	SI-022B	CDF and LERF values decreased	No change	No change	See Comment for SI-022B.
SI	SI-023B	CDF and LERF values decreased	No change	No change	See Comment for SI-022B.
SI	SI-024B	CDF and LERF values decreased	No change	No change	See Comment for SI-022B.
SI	SI-060	CDF and LERF values decreased	No change	No change	See Comment for SI-022B.
SI	SI-088	CDF and LERF values decreased	No change	Changed from 18 months to continuous	Test interval change reduced failure probability used in risk evaluation.
SQ	None				

Table 3-2 SQN Unit 2 Segments that Changed From HSS to LSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
AF	None				
BD	BD-005	CDF and LERF values decreased	No change	No change	Improvement in the reliability and availability of equipment used to mitigate secondary side breaks (AFW) resulted in a decrease in CDF and LERF.
BD	BD-006	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-007	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-008	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-010	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-011	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-018	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
BD	BD-019	CDF and LERF values decreased	No change	No change	See Comment for BD-005.
CH	CH-018	CDF and LERF values decreased	No change	No change	Changes to PRA results after the initial expert panel meetings resulted in RRWs that supported a LSS ranking. Due to timing considerations, the expert panel chose to leave this segment as HSS for the original submittal. Revised inputs support LSS ranking.
CH	CH-019	CDF and LERF values decreased	No change	No change	See Comment for CH-018.
CH	CH-020	CDF and LERF values decreased	No change	No change	See Comment for CH-018.
CI	None				
CS	None				

Table 3-2 SQN Unit 2 Segments that Changed From HSS to LSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
FW	FW-005	LERF values decreased	No change	No change	Final RRWs for original submittal were less than 1.005, but the segment was ranked HSS. Segment was originally HSS because of LERF results. LERF results decreased primarily because of an updated operator action (backup to actuation signals) in the model. The HRA calculator was used in the first major update to the HRA since 1992.
FW	FW-006	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-007	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-008	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-009	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-010	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-011	LERF values decreased	No change	No change	See Comment for FW-005.
FW	FW-012	LERF values decreased	No change	No change	See Comment for FW-005.
MS	None				
RC	RC-017	CDF and LERF values decreased	No change	No change	Final RRWs for original submittal were less than 1.005, but the segment was ranked HSS. Primary contributor to the change is the recalculation of the operator action failure probability for aligning high pressure recirculation. The HRA calculator was used in the first major update to the HRA since 1992.
RC	RC-018	CDF and LERF values decreased	No change	No change	See Comment for RC-017.
RC	RC-020	CDF and LERF values decreased	No change	No change	See Comment for RC-017.
RC	RC-025	CDF and LERF values decreased	No change	No change	See Comment for RC-017.

Table 3-2 SQN Unit 2 Segments that Changed From HSS to LSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
RC	RC-027	CDF and LERF values decreased	No change	No change	See Comment for RC-017.
RC	RC-028	CDF and LERF values decreased	No change	No change	See Comment for RC-017.
RC	RC-029	CDF and LERF values decreased	Decreased	No change	See Comment for RC-017. In addition, the failure probability decreased because the "with ISI" failure probability was used. The segment was added to the augmented inspection program for Alloy 600.
RC	RC-047	CDF and LERF values decreased	No significant change	No change	See Comment for RC-017.
RC	RC-048	CDF and LERF values decreased	No significant change	No change	See Comment for RC-017.
RC	RC-049	CDF and LERF values decreased	No significant change	No change	See Comment for RC-017.
RC	RC-051	CDF and LERF values decreased.	No change	No change	See Comment for RC-017.
RH	None				
SI	SI-021B	CDF and LERF values decreased	Increased	No change	Changes to PRA results after the initial expert panel meetings resulted in RRWs that supported a LSS ranking. Due to timing considerations, the expert panel chose to leave this segment as HSS for the original submittal. Revised inputs support LSS ranking.
SI	SI-022B	CDF and LERF values decreased	No change	No change	See Comment for SI-022B.
SI	SI-023B	CDF and LERF values decreased	No change	No change	See Comment for SI-022B.
SI	SI-024B	CDF and LERF values decreased	No change	No change	See Comment for SI-022B.
SI	SI-060	CDF and LERF values decreased	No change	No change	See Comment for SI-022B.
SI	SI-088	CDF and LERF values decreased	No change	Changed from 18 months to continuous	Test interval change reduced failure probability used in risk evaluation.
SQ	None				

Table 3-3 SQN Unit 1 Segments that Changed From LSS to HSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
AF	AF-037	CDF and LERF values increased	No change	No change	PRA results conservatively account for a loss of TDAFWP and a loss of SG ARVs due to environmental conditions resulting from a break in this segment.
AF	AF-038	CDF and LERF values increased	No change	No change	See Comment for AF-037
AF	AF-039	CDF and LERF values increased	No change	No change	See Comment for AF-037
AF	AF-041	CDF and LERF values increased	No change	No change	PRA results conservatively account for a loss of SG ARVs due to environmental conditions resulting from a break in this segment.
BD	BD-021	CDF and LERF values increased	No change	No change	Only the RRW for LERF without operator action is greater than 1.005. The total plant piping LERF without operator action decreased from the value for the original submittal and the segment LERF without operator action increased resulting in the high RRW.
BD	BD-024	CDF and LERF values increased	No change	No change	See Comment for BD-021.
CH	None				
CI	None				
CS	CS-005	LERF values increased	No change	No change	LERF set equal to CDF because failure of this segment can lead to loss of containment sump and loss of containment isolation boundary.
CS	CS-006	LERF values increased	No change	No change	See Comment for CS-005.
FW	None				
MS	MS-017	CDF values increased	No change	No change	Revised PRA model includes induced steam generator tube rupture due to steamline break resulting in increased consequences. RRWs for CDF are greater than 1.005.
MS	MS-018	CDF values increased	No change	No change	See Comment for MS-017.

Table 3-3 SQN Unit 1 Segments that Changed From LSS to HSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
MS	MS-019	CDF values increased	No change	No change	See Comment for MS-017.
MS	MS-020	CDF values increased	No change	No change	See Comment for MS-017.
RC	None				
RH	None				
SI	SI-047C	New segment	New segment	New segment	A new segment was created from an existing segment. The new segment has a smaller weld thickness and a different pipe size than the remaining portion of the original segment.
SI	SI-081B	CDF and LERF values decreased	Increased	Changed from refueling to monthly	Based on industry experience, the potential for thermal striping, vibration, and stress corrosion cracking are modeled in revised failure probabilities. Increase in failure probability resulted in the increased RRWs.
SI	SI-089	CDF and LERF values decreased	No change	No change	All RRWs are less than 1.005; however, the expert panel ranked the segment HSS due to the LERF with operator action RRW. The total plant piping LERF with operator action decreased from the value for the original submittal. This increased the segment RRW. The segment was evaluated in conjunction with SI-090.
SI	SI-090	CDF and LERF values decreased	No change	No change	All RRWs are less than 1.005; however, the expert panel ranked the segment HSS due to the LERF RRWs. The total plant piping LERF cases for without and with operator action decreased from the values for the original submittal. This increased the segment RRWs. The segment was evaluated in conjunction with SI-089.
SQ	None				

Table 3-4 SQN Unit 2 Segments that Changed From LSS to HSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
AF	AF-037	CDF and LERF values increased	No change	No change	PRA results conservatively account for a loss of TDAFWP and a loss of SG ARVs due to environmental conditions resulting from a break in this segment.
AF	AF-038	CDF and LERF values increased	No change	No change	See Comment for AF-037
AF	AF-039	CDF and LERF values increased	No change	No change	See Comment for AF-037
AF	AF-041	CDF and LERF values increased	No change	No change	PRA results conservatively account for a loss of SG ARVs due to environmental conditions resulting from a break in this segment.
BD	BD-021	CDF and LERF values increased	No change	No change	All RRWs are less than 1.005, however, the expert panel ranked the segment HSS based on the RRW for LERF without operator action. The total plant piping LERF without operator action decreased from the value for the original submittal and the segment LERF without operator action increased resulting in the high RRW.
BD	BD-024	CDF and LERF values increased	No change	No change	See Comment for BD-021.
CH	None				
CI	None				
CS	CS-005	LERF values increased	No change	No change	LERF set equal to CDF because failure of this segment can lead to loss of containment sump and loss of containment isolation boundary.
CS	CS-006	LERF values increased	No change	No change	See Comment for CS-005.
FW	None				
MS	MS-017	CDF values increased	No change	No change	Revised PRA model includes induced steam generator tube rupture due to steamline break resulting in increased consequences. RRWs for CDF are greater than 1.005.
MS	MS-018	CDF values increased	No change	No change	See Comment for MS-017.

Table 3-4 SQN Unit 2 Segments that Changed From LSS to HSS					
System ID	Segment ID	PRA Results	Failure Probability Results	Test Interval	Comments
MS	MS-019	CDF values increased	No change	No change	See Comment for MS-017.
MS	MS-020	CDF values increased	No change	No change	See Comment for MS-017.
RC	None				
RH	None				
SI	SI-081B	CDF and LERF values decreased	Increased	Changed from refueling to monthly	Based on industry experience, the potential for thermal striping, vibration, and stress corrosion cracking are modeled in revised failure probabilities. Increase in failure probability resulted in the increased RRWs.
SI	SI-089	CDF and LERF values decreased	No change	No change	All RRWs are less than 1.005, however, the expert panel ranked the segment HSS due to the LERF with operator action RRW. The total plant piping LERF with operator action decreased from the value for the original submittal. This increased the segment RRW. The segment was evaluated in conjunction with SI-090.
SI	SI-090	CDF and LERF values decreased	No change	No change	All RRWs are less than 1.005; however, the expert panel ranked the segment HSS due to the LERF RRWs. The total plant piping LERF cases for without and with operator action decreased from the values for the original submittal. This increased the segment RRWs. The segment was evaluated in conjunction with SI-089.
SQ	None				

NRC Question 3 (continued)

b. **For the multiple pipe diameter (MPD) segments, have their failure probability (FP) changes been revised as a result of using WCAP-14572 Supplement 2? If so, describe how you implemented Supplement 2 methodology.**

TVA Response

The method used for calculating the failure probabilities for multiple pipe size segments in the original RI-ISI submittal is consistent with WCAP-14572, Supplement 2. Therefore, failure probabilities were not recalculated for the update as a result of using WCAP-14572, Supplement 2.

c. **How were the number of examinations for these MDP segments determined and has the number or the location of these examinations been revised as a result of using WCAP-14572 Supplement 2?**

TVA Response

The number of examinations for the HSS multiple pipe size segments were determined using the process described in Section 3.7 of WCAP-14572. The requirements in WCAP-14572, Supplement 2, were also reviewed and the final number of examinations meets the requirements of Supplement 2.

d. **Identify and provide expert panel (EP) justification for those pipe segments with revised RRW ≥ 1.005 that were reclassified by the EP decision.**

TVA Response

Four segments for each unit had one or more RRWs greater than 1.005, but were ranked LSS by the expert panel. The segments are AF-023, AF-024, CS-007, and CS-008 for both units.

The following is a summary of the expert panel discussion for segments AF-023 and AF-024. A pipe break in these segments post-accident would be identified by the control room operators via:

1) Pipe break indication that specifically identifies either loop #2 or #3 that will illuminate at less than

100 pounds per square inch absolute (psia) on the main control board 1(2)-M-3.

2) High flow indication for the failed loop with a reduced flow on the other three loops on the main control board 1(2)-M-4.

3) Steam generator level will not increase on the failed loop while other loops' increase will be slow to recover.

4) The time available to identify and isolate the leaking AFW supply line depends upon the initiating accident and the number of AFW pumps available. The most likely scenario is a reactor trip with both motor driven auxiliary feedwater pumps (MDAFWPs) available. In this case, for the limiting leak, two steam generators (SGs)do not receive any AFW until the leaking line is isolated; however, decay heat is removed through the two unaffected SGs with AFW supplied by the unaffected MDAFWP so there is no time requirement to isolate the leaking line.

5) Identification of the leaking AFW line is performed in E-0, *Reactor Trip or Safety Injection*, and isolating the leaking line is performed in E-2, *Faulted Steam Generator Isolation*.

The following is a summary of the expert panel discussion for segments CS-007 and CS-008. The RHR spray headers are normally isolated and are manually placed in service. Operator action is creditable for the following reasons:

- Sufficient time is available since operation of the residual heat removal (RHR) spray occurs late in the event after all containment ice has melted.

- Sufficient sump level indication exists and would be monitored by SQN's Technical Support Center at this point in the event to detect loss of containment sump inventory and increase in Auxiliary Building passive sump. Operators would be monitoring containment pressure and detect continued increase after RHR spray initiation.

- RHR header isolation valves FCV-72-40 and -41 are environmentally qualified and receive emergency power and would be operable to isolate pipe breaks in the annulus.

- Training would cause the operator to evaluate and take action to limit containment sump loss.

- The most likely scenario, that would result in RHR spray being placed in operation, is a large break loss-of-coolant accident (LBLOCA) with successful sump recirculation. In this case, a leaking RHR spray line results in a decrease in sump level. The volume of water available in the sump following a LBLOCA is >500,000 gallons with <300,000 gallons needed for sump recirculation so at least an hour is available to identify and isolate a leaking RHR spray line.

e. Describe how the risk impacts of external events, internal fire, and shutdown were considered and evaluated by the EP.

TVA Response:

The risk impacts of external events, internal fire, and shutdown were considered and evaluated by the expert panel in accordance with WCAP-14572, Section 3.6.2 for the original RI-ISI program, and that information was reviewed and included for the expert panel's consideration for the update.

NRC Question 4

Has the uncertainty analysis as discussed on page 125 of the WCAP-14572 been re-performed? If so, please identify those systems/components for which the RRW increased to or above 1.005 and provide the results of the EP's evaluation of these segments. If not, provide a description and justification of how your process considered uncertainty and why the deviation is acceptable?

TVA Response

The uncertainty analysis was performed integral to the risk analysis and the results with uncertainty were reviewed by the

expert panel. By addressing the uncertainty in this manner, a list of segments whose RRWs became greater than 1.005 when the uncertainty was added does not exist. This method of addressing uncertainty is identical to that used in the original RI-ISI program and is described in response to NRC RAI No. 4 on the original RI-ISI submittal (refer to TVA letter to NRC dated August 31, 2001, Docket Nos. 50-327 and 50-328). The treatment of uncertainty was approved for the original program as stated in
Section 4 of the safety evaluation transmitted in a NRC letter dated October 19, 2001 (TAC Nos. MB1566 and MB1567).

NRC Question 5

For Unit 1, based on a comparison between Table 5-1 on page E-19 of the previous submittal and Table 1 on page E1-5 of the current submittal, the following information and/or explanations are requested.

a. **Page E1-2 of the current submittal, last paragraph indicates that 16 LSS segments were reclassified as HSS and 36 HSS segments were reclassified as LSS segments for Unit 1. However, staff review of the above tables seems to indicate that only 12 LSS segments were reclassified as HSS and 30 HSS segments were reclassified as LSS segments. Please explain the difference in these numbers.**

   TVA Response

   It is not possible to determine the total number of segments that change from HSS to LSS (or LSS to HSS) based on comparing the total system HSS segments in Table 5-1 on page E-19 of the previous submittal and Table 1 on page E1-5 of the current submittal. This is because the information presented reflects the total number of HSS segments for each system. As an example, the number of system BD HSS segments has decreased from 12 to 6 in the two tables. The decrease occurred because 8 BD segments changed from HSS to LSS and 2 BD segments changed from LSS to HSS. Note that the same reduction of 6 HSS segments also could have occurred if 6 BD segments changed from HSS to LSS and 0 BD segments changed from LSS to HSS.

b. **Was there a reduction in the number of inspection locations in any pipe segment that was and remains a HSS segment? If so, how great was the reduction and why was the reduction taken.**

TVA Response

For the segments that were HSS in the second interval and also HSS for the third interval, there are no reductions in the number of examination locations.

c. **On page E1-7, Code Category C-F-1, 3rd Interval RI-ISI, it appears that the total numbers of exam locations should be 27 and 8 for nondestructive examination (NDE) and VT2 (not 25 and 10), respectively. Please correct these numbers or otherwise explain this apparent discrepancy.**

TVA Response

The number of examination locations per system is summarized in Table 5.d below for American Society of Mechanical Engineers (ASME) Code Category C-F-1, 3rd Interval RI-ISI. This is consistent with the sum shown on page E1-7.

Table 5.d Sequoyah Unit 1 C-F-1 Examinations		
System	# NDE Locations	# VT-2 Examinations
AF	0	0
BD	0	0
CH	0	2
CI	0	0
CS	5	1
FW	0	0
MS	0	0
RC	0	0
RH	6	1
SI	14	6
SQ	0	0
Total	25	10

d. **Please explain how and why you have chosen two additional examinations of reactor coolant system to meet the change in risk criteria.**

TVA Response

The guidance for evaluating and meeting the change in risk criteria in WCAP-14572 was followed. Examinations were added to two segments in the Reactor Coolant System to meet the change in risk criteria.

NRC Question 6

For Unit 2, based on a comparison between Table 5-2 on page E-21 of the previous submittal and Table 2 on page E1-8 of the current submittal, the following information and/or explanations are requested.

a. **Page E1-2, last paragraph indicates that 15 LSS segments were reclassified as HSS and 36 HSS segments were reclassified as LSS segments for Unit 2. However, staff review of the above tables seems to indicate that Table 5-2 shows only 10 LSS segments that were reclassified as HSS and 31 HSS segments that were reclassified as LSS segments. Please explain the difference in these numbers.**

   TVA Response

   Refer to the response to NRC Question 5.a. above.

b. **Was there a reduction in the number of inspection locations in any pipe segment that was and remains a HSS segment? If so, how great was the reduction an why was the reduction taken.**

   TVA Response

   The information is provided in Table 3-2 above.

c. **On page E1-10, Code Category C-F-1, 3rd Interval RI-ISI, it appears that the total numbers of exam locations are 24 and 8 for NDE and VT2 (not 22 and 10), respectively. Please correct these numbers or otherwise explain this apparent discrepancy.**

   TVA Response

   The number of examination locations per system is summarized in Table 6.d below for ASME Code Category C-F-1, 3rd Interval RI-ISI. This is consistent with the sum shown on page E1-10.

Table 6.d SQN Unit 1 C-F-1 Examinations		
System	# NDE Locations	# VT-2 Examinations
AF	0	0
BD	0	0
CH	0	2
CI	0	0
CS	5	1
FW	0	0
MS	0	0
RC	0	0
RH	4	1
SI	13	6
SQ	0	0
Total	22	10

d. Please explain how and why you have chosen one additional examination of Reactor Coolant system to meet the change in risk criteria.

TVA Response

The guidance for evaluating and meeting the change in risk criteria in WCAP-14572 was followed. One additional examination was added to one segment in the Reactor Coolant System to meet the change in risk criteria.

NRC Question 7

The newer versions of the American Society of Mechanical Engineers (ASME) Code have reduced the exempted portions of auxiliary feedwater piping from NPS 4 to NPS 1 1/2. This reduction in exempted piping has caused other licensees to add ASME Class 2 and/or Class 3 Auxiliary Feedwater piping to the scope of their RI-IS1 programs, and to implement the (EPRI or) WCAP-14572 methodology to classify, risk-rank, and to select, as necessary, additional locations for the next ISI interval. Please describe how you treated this issue.

TVA Response

For the original SQN Units 1 and 2 RI-ISI programs, all piping within the Class 1 and 2 boundaries was included in the analysis even if it was exempt from inspection by ASME Section XI. This was not changed for the updated analysis. Changes to the Section XI exemptions do not have an impact on the SQN Units 1 and 2 RI-ISI programs.

NRC Question 8

Based on industry experience with primary water stress corrosion cracking (PWSCC) of Alloy 600 and associated weld material Alloy 82/182 the staff finds that these welds should be inspected at a minimum in accordance with ASME Code Section XI requirements until such time that industry has gathered sufficient data to justify incorporating these items into the RI-ISI program.

Please describe what examinations (e.g., UT-Appendix VIII) will be performed and at what frequency for each dissimilar metal welds containing either Alloy 600 or its associated weld metal in Class 1 and 2 components for each unit.

TVA Response

Examinations for Alloy 600 and associated weld material Alloy 82/182 will be conducted in accordance with the current EPRI MRP guidelines.

NRC Question 9

In Enclosure 1, Tables 1 and 2 of the April 21, 2006, submittal, what is meant by footnote c. Augmented programs for erosion-corrosion (including MIC) continue.? How will the NDE examinations for these elements be performed (Appendix VIII, thickness measurements or other)?

TVA Response

Footnote c means that the RI-ISI program will not change the current augmented examination programs at both units and the examinations will continue to be performed in accordance with the augmented programs. This is the same approach taken in the original RI-ISI submittal.

NRC Question 10

In Enclosure 1, Tables 1 and 2, please describe what is meant by footnotes o and p.

TVA Response

Footnotes o and p mean that the total number of examinations required to be performed for the RI-ISI program include examinations that are being performed at the plants as part of

their augmented programs. These augmented program examinations satisfy the requirement for examination for cause in some of the high safety-significant segments.

NRC Question 11

Provide justification on how adding VT-2 examinations will maintain risk neutrality, since VT-2 examinations are already required (footnote a).

TVA Response

Adding VT-2 examinations helps to maintain risk neutrality by the examinations being performed at a more frequent interval than required by the ASME Code. VT-2 examinations required as part of the RI-ISI program are performed every refueling outage.

NRC Question 12

For footnotes a, b, d, e, g, and l, what is the frequency of the VT-2 visual examinations?

TVA Response

Footnote a indicates that the system pressure tests and VT-2 examinations required by ASME Section XI will continue for Class 1, 2, and 3 systems as required by Section XI. These tests are performed in accordance with the Section XI requirements. Footnotes b, d, e, g, and l are for VT-2 examinations that are part of the RI-ISI program and are conducted every refueling outage.

NRC Question 13

For each NDE examination list what type of NDE examination will be performed (e.g., UT-Appendix VIII, UT-Thickness, surface examination).

TVA Response

The type of NDE examination that will be performed is determined by Table 4.1-1 of WCAP-14572, Revision 1-NP-A, Supplement 2. Note that WCAP-14572, Revision 1-NP-A and WCAP-14572, Revision 1-NP-A, Supplement 2 do not specify whether UT volumetric examinations are performed in accordance with Appendix VIII or Section V of the ASME code.

TVA utilizes the following ASME code criteria for NDE examinations:

> Ultrasonic examinations that are performed as part of the RI-ISI program are performed in accordance with ASME code, Section XI, Appendix VIII. Surface examinations and visual examinations are performed in accordance with the applicable portions of ASME code, Section XI and Section V.

NRC Question 14

Discuss the changes to the feedwater system that eliminated the examinations that were required in the 2nd ISI interval and no longer required in the 3rd ISI interval.

TVA Response

The information is provided in Tables 3-1 and 3-2 above.