April 30, 2007

Mr. Preston D. Swafford
Interim Chief Nuclear Officer
Nuclear Support
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 — RISK-INFORMED INSERVICE INSPECTION PROGRAM FOR THE THIRD 10-YEAR INTERVALS (TAC NOS. MD1452, MD1453, MD1454 AND MD1455)

Dear Mr. Swafford:

By letter dated April 21, 2006, as supplemented by letter dated April 16, 2006, the Tennessee Valley Authority submitted Relief Requests (RRs) 1-RI-ISI-1 and 1-RI-ISI-2 for Sequoyah Nuclear Plant (SQN) Unit 1, and RRs 2-RI-ISI-1 and 2-RI-ISI-2 for SQN Unit 2.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i), RRs 1-RI-ISI-1 and 2-RI-ISI-1 propose an alternative to the inspection program requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI. Specifically, these RRs request to extend the SQN Risk-Informed Inservice Inspection (RI-ISI) Program Plan to the third 10-year ISI intervals at SQN Units 1 and 2. The SQN RI-ISI program was previously approved by the Nuclear Regulatory Commission for use in the second 10-year ISI interval at each unit. Based on its review of your submittals, the U.S. Nuclear Regulatory Commission (NRC) staff concluded that the alternatives proposed in RRs 1-RI-ISI-1 and 2-RI-ISI-1 provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), they are authorized for the third 10-year ISI intervals at SQN Units 1 and 2, which began June 1, 2006.

Pursuant to 10 CFR 50.55a(a)(3)(ii), RRs 1-RI-ISI-2 and 2-RI-ISI-2 propose visual examination of certain welds in lieu of the RI-ISI program requirement to perform volumetric examination. Based on its review of your submittals, the NRC staff concluded that compliance with the specified requirements would result in hardship or difficulty without a compensating increase in the level of quality and safety and the proposed alternative provides reasonable assurance of structural integrity of the affected components. Therefore, RRs 1-RI-ISI-2 and 2-RI-ISI-2 are authorized, pursuant to 10 CFR 50.55a(a)(3)(ii), for the third 10-year ISI intervals at SQN Units 1 and 2, which began June 1, 2006.

If there are any questions regarding this matter, please contact the SQN Project Manager, Brendan T. Moroney, at 301-216-2778.

Sincerely,

/RA/

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosure: Safety Evaluation

cc w/enclosure: See next page

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If there are any questions regarding this matter, please contact the SQN Project Manager, Brendan T. Moroney, at 301-216-2778.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NOS. 1-RI-ISI-1, 2-RI-ISI-1, 1-RI-ISI-2 AND 2-RI-ISI-2

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By letter dated April 21, 2006 (Reference 1, the submittal), as supplemented by letter dated April 16, 2007 (Reference 2), the Tennessee Valley Authority (TVA, the licensee) submitted relief requests (RRs) for the third 10-year Inservice Inspection interval at Sequoyah Nuclear Plant (SQN) Units 1 and 2. RRs 1-RI-ISI-1and 2-RI-ISI-1 propose to extend the Risk-Informed Inservice Inspection (RI-ISI) Program Plan for SQN Units 1 and 2 to the third 10-year ISI interval. The SQN Units 1 and 2 RI-ISI programs were initially submitted to the U.S. Nuclear Regulatory Commission (NRC) in a letter dated March 23, 2001 (Reference 3), during the second 10-year ISI interval, and supplemented in letters dated August 31, 2001 and October 3, 2001 (References 5 and 6). The SQN RI-ISI program was reviewed and approved by the NRC for use in the second 10-year ISI interval in a letter dated October 19, 2001 (Reference 7), as clarified by letter dated October 23, 2001 (Reference 8). RRs 1-RI-ISI-2 and 2-RI-ISI-2 propose visual examination of certain welds in lieu of the RI-ISI program requirement to perform volumetric examination.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.55a(g) specifies that ISI of nuclear power plant components shall be performed in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The licensee's RI-ISI program, as outlined in References 3, 5, and 6, was developed in accordance with the methodology contained in the Westinghouse Owners Group (WOG) report WCAP-14572, Revision 1-NP-A, (Reference 9, the topical) which was reviewed and approved by the NRC staff (the staff), as documented in Reference 10. The SQN RI-ISI program is an alternative pursuant to 10 CFR 50.55a(a)(3)(i). In Reference 1, the licensee requests NRC authorization to extend its RI-ISI program, previously approved for use in the second interval, for use in the third ISI interval at SQN Units 1 and 2. The scope of the SQN RI-ISI program is limited to ASME Code Class 1 and Class 2 piping welds.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for ISI of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ISI Code of record for the third 10-year ISI interval for Sequoyah Units 1 and 2 is the 2001 Edition through the 2003 Addenda of the ASME Section XI Code.

3.0 TECHNICAL EVALUATION FOR 1-RI-ISI-1 AND 2-RI-ISI-1

The licensee is requesting relief for continued use of the approved RI-ISI program plan instead of the ASME Code Section XI program in the third 10-year ISI interval. An acceptable RI-ISI program plan is expected to meet the five key principles of risk-informed decisionmaking, discussed in Regulatory Guide (RG) 1.178, Revision 1, “An Approach for Plant-Specific Risk-Informed Decision Making: Inservice Inspection of Piping” (Reference 15), Standard Review Plan 3.9.8 (Reference 16), NUREG-0800 Chapter 19 (Reference 18), and the WOG Topical Report, WCAP-14572, Revision 1-NP-A, Revision 1-NP-A Supplement 1, and Revision 1-NP-A Supplement 2 (References 9 and 11), as stated below.

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency (CDF) or risk, the increases should be small and consistent with the intent of the Commission’s Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored by using performance measurement strategies.

The first principle is met because an alternative ISI program may be authorized pursuant to 10 CFR 50.55a(3) and, therefore, an exemption request is not required.

The second and third principles require assurance that the alternative program is consistent with the defense-in-depth philosophy and that sufficient safety margins are maintained, respectively. The methodology used to develop the SQN third 10-year RI-ISI program interval is unchanged from the methodology approved for use in the SQN second 10-year RI-ISI interval program.

In Reference 1, the licensee states that the SQN RI-ISI program for the third inspection interval is consistent with the SQN second RI-ISI interval and includes changes from WCAP-14572, Supplement 2 (Reference 11). The process used in the submittal to develop the proposed third

RI-ISI interval program is the same as that outlined in the TVA's original submittal of Reference 3, which was approved by the NRC in Reference 7. In accordance with Section 3.6.2 of WCAP-14572, Revision 1-NP-A, an expert panel evaluated the risk-informed results and made a final decision by identifying the safety significance of each piping segment. The licensee also states in Reference 2 that the risk impacts of external events, internal fire events, and shutdown were evaluated and included for the expert panel's consideration for reclassifying the safety significance of segments. An uncertainty analysis was performed integral to the risk analysis and the results with uncertainty were reviewed by the expert panel, instead of performing a separate analysis as discussed in WCAP-14572. The method of addressing uncertainty is identical to that used in the original RI-ISI program described in References 3 and 5, which was approved by the NRC in Reference 7.

In the submittal, the licensee provides an Enclosure 1 with two tables (Table 1 for Unit 1 and Table 2 for Unit 2) comparing the existing RI-ISI program (i.e. - the number of nondestructive examinations (NDEs) within each system in scope) for the second interval with the proposed RI-ISI program for the third interval. For Unit 1, the licensee proposes reducing the numbers of NDEs and visual examinations (VT2) from 91 and 64 in the second interval to 68 and 44 in the third interval, respectively. For Unit 2, the licensee proposes reducing the numbers of NDEs and VT2 from 85 and 66 in the second interval to 62 and 43 in the third interval, respectively. These reductions are the result of the reclassification of pipe segments based on the revised segment failure probabilities, updated test interval, and/or revised consequences using the SQN probabilistic risk assessment (PRA) Revision 3 model. The licensee also notes that the eight reactor coolant system examinations added for defense-in-depth considerations for each unit in the previously approved RI-ISI program for SQN's second inspection interval remain the same.

A comparison with SQN Unit 1 second 10-year interval RI-ISI indicates that 16 low safety significant (LSS) segments were reclassified as high safety significant (HSS) and 36 HSS segments were reclassified as LSS. A comparison with SQN Unit 2 indicates that 15 LSS segments were reclassified as HSS and 36 HSS segments were reclassified as LSS segments. These reclassifications and existing segment classifications were reviewed and concurred on by the SQN RI-ISI expert panel.

In Reference 2, the licensee provides additional detail on how the pipe segment reclassifications were arrived at. The licensee shows that the RI-ISI consequence evaluation process was repeated using SQN PRA Revision 3, resulting in decreased pipe segment CDF and larege early release frequency (LERF) values and risk reduction worth (RRW) values. In some cases, the safety significance was reduced from HSS to LSS due to the decrease in pipe failure probability or change in test interval.

The licensee also indicates that there were four segments for each unit that had one or more RRWs greater than 1.005, but were ranked LSS by the expert panel. These segments are AF-023, AF-024, CS-007 and CS-008 for both units. The SQN expert panel rationales and justifications for segments AF-023 and AF-024 are as follows:

- There are control room indications that enable the control room operators to identify a pipe break in these segments post-accident. These include: (1) pipe break indication that specifically identifies a low pressure in either loop #2 or #3 and (2) high flow indication for the failed loop with a reduced flow on the other three loops.

- The steam generator level will not increase on the failed loop and will be slow to recover in the other loops.
- The time available to identify and isolate the leaking Auxiliary Feedwater (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 AFW pumps available. In this case, for the limiting leak, two steam generators do not receive any AFW until the leaking line is isolated; however, decay heat is removed through the two unaffected steam generators with AFW supplied by the unaffected motor driven AFW pump so there is no time requirement to isolate the leaking line.
- Plant procedures include instructions for identifying and isolating a leaking AFW line.

The following is a summary of the expert panel justification for segments CS-007 and CS-008. The Residual Heat Removal (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 RHR spray occurs late in the event after melting of all containment ice.
- Sufficient sump level indication exists and would be monitored to detect loss of containment sump inventory and increase in Auxiliary Building passive sump level. Operators would be monitoring containment pressure and would 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 a piping break in the annulus.
- Operators are trained 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 LOCA (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.

Based on the above discussion and additional information provided in References 1 and 2, the NRC staff concluded that the licensee's methodology for evaluating and developing the second 10-year interval RI-ISI program has been appropriately reapplied in updating this program for the third 10-year interval. The method used for reclassifying the safety significance of segments is consistent the WCAP-14572, Supplement 2. Hence, the proposed changes are consistent with the defense-in-depth philosophy and maintain safety margins. Therefore, the staff concludes that the second and third key principles have been met.

The fourth principle requires an estimate of the change in risk, and the change in risk is dependent on the location of inspections in the proposed ISI program compared to the location of inspections that would be inspected using the requirements of ASME Code, Section XI. The topical requires that a change in risk measurement must consider the discontinuance of ASME Code-required inspections, as well as any new inspections resulting from the application of its methodology. Reference 3 indicated that the code of record for the SQN second 10-year ISI interval is the 1989 Edition of Section XI of the ASME Code. For the third interval, the licensee plans to update to the 2001 Edition, 2003 Addenda of Section XI of the ASME Code. It is possible that, if a revised ASME Code inspection program for the third interval was to be developed for this updated code of record, the number and/or locations of inspections mandated by the updated code of record could be increased because of possible changes in code sampling percentage requirements. However, development of an acceptable RI-ISI program is primarily achieved through the risk-ranking and the inspection location selection processes. When applied as part of an integrated decision-making process, subsequent change in risk estimates provide reasonable assurance that the change in the ISI program would result in a total plant risk neutrality, risk decrease, or a small risk increase and that will be consistent with staff guidelines found in Regulatory Guide 1.174 (Reference 14).

The PRA inputs reported in the SQN third interval RI-ISI relief request to calculate the change in risk are derived from the SQN PRA Revision 3 model, dated August, 2004. The licensee states in Reference 2 that all of the Level A Facts and Observations (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 PRA model. The staff believes that the SQN Revision 3 PRA is of higher quality than the SQN Revision 1 PRA model used for the original RI-ISI relief request. The licensee also states in Reference 2 that the results of the consequence analysis for the third 10-year interval RI-ISI demonstrate that the relative importance of pipe segment failures behaves as expected. For example,

- Pipe segment failures that cause the loss of a train of a risk-significant system result in a larger increase in CDF and LERF compared to other systems.
- Pipe segment failures that cause the loss of both trains of a system or cause the loss of multiple 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 results in a relatively large increase in LERF compared to those pipe segments whose failure does not affect containment performance.

Based on the discussion above, the staff concludes that the SQN Revision 3 PRA Model is of sufficient quality to be used in support of the current relief request. This is consistent with Section 4.5.2 of the topical which notes that the PRA used in the development of any RI-ISI program is a state of knowledge at the time of implementation and takes credit for the existence of the monitoring and feedback program.

The licensee states in the submittal and Reference 2 that a new delta risk evaluation was performed following the guidance for evaluating and meeting the change in risk criteria in WCAP-14572. The change in risk evaluation was reperformed to compare the original Section XI programs with the revised third inspection interval RI-ISI program for each unit. Two reactor coolant system segments were required to be added to the SQN Unit 1 RI-ISI program to meet the change in risk criteria of WCAP-14572. For SQN Unit 2, one reactor coolant system segment was identified to be added to the RI-ISI program to meet the change in risk criteria. The baseline CDF from this model was calculated to be 1.31E-5 per year. The baseline LERF was calculated to be 2.62E-7 per year. These are less than the NRC safety goal of less than 1E-4 per year for CDF and less than 1E-5 per year for LERF. The updated RI-ISI analysis indicates a slight reduction in risk, hence the third interval SQN RI-ISI program is considered to be risk neutral. Therefore, the calculated values are acceptable.

Given the above considerations concerning the increase in risk and SQN PRA quality, the staff finds that the licensee's analysis provides assurance that the fourth key principle is met, thus, implementation of the proposed program will not cause the NRC safety goals to be exceeded.

For the fifth principle of risk-informed decisionmaking, Section 4.5.2 of WCAP-14752 states that RI-ISI programs are living programs and should be monitored continuously, and that monitoring of these programs encompasses many facets of feedback or corrective action which includes periodic updates. As stated in the submittal (Reference 1), the SQN RI-ISI program is a “living” program and the information has been updated and analyzed in accordance with the WOG Topical Report, WCAP-14572, Revision 1-NP-A, Revision 1-NP-A Supplement 1, and Revision 1-NP-A Supplement 2. Consistent with these topical reports, new information has been incorporated in the SQN RI-ISI analysis as part of the “living” RI-ISI program. This includes revised consequences for pipe segments based on SQN PRA Revision 3, revised failure probabilities for a limited number of segments based on industry and plant experience and plant modifications, and updated test intervals for a limited number of segments. The revised locations for inspection have been selected in accordance with the approved methodology of Reference 9. In Reference 2, the licensee provided four tables indicating the updates in PRA result, segment failure probability, and test interval. Based on its review, the NRC staff concludes that the licensee’s RI-ISI program is consistent with the “living program” concept, and therefore, the fifth key principle is met.

Due to recent and ongoing issues related to degradation due to primary water stress corrosion cracking in components that contain alloy 600/82/182, the staff requested that the licensee provide information related to welds containing alloy 82/182. The licensee responded that it will comply with the selection criteria and frequency of the Electric Power Research Institute Materials Reliability Program (MRP) contained in “Primary System Piping Butt Weld Inspection and Evaluation Guideline (MRP-139).” The staff is also aware of additional commitments regarding alloy 600/82/182 butt welds in pressurizer nozzles at SQN Unit 1(Reference 19). These commitments address current NRC staff concerns regarding this issue and are acceptable.

NDE requirements, including qualification, examination coverage calculation, detection and sizing used to perform the examinations on the items selected for examination will be in accordance with the 2001 Edition through the 2003 Addenda of the ASME Code, Section XI.

Based on the above discussion, the staff concludes that the five key principles of risk-informed decision making are ensured by the licensee’s proposed third 10-year RI-ISI program, and therefore the proposed program for the third 10-year ISI interval is acceptable.

4.0 TECHNICAL EVALUATION FOR 1-RI-ISI-2 AND 2-RI-ISI-2

4.1 Components for Which Relief is Requested

- ASME Code Class 1 and 2 socket welds and branch connections nominal pipe size (NPS) 2 inches and smaller.
- In addition, two steam generator blowdown system segments at each unit and two containment spray system segments on each unit.

4.2 Requirement from Which Relief is Request

Relief is requested from performing a volumetric examination of socket welds and branch connection welds less than or equal to 2 inches NPS subject to thermal fatigue. Relief is also requested from performing volumetric examination on two steam generator blowdown system piping welds located inside containment penetration sleeves at each unit. In addition, relief is requested from performing VT-2 examinations for a portion of two piping segments in the containment spray system that are open ended. These requirements are based on the ASME Code Section XI requirements as modified by the licensee's RI-ISI program contained in RRs 1-RI-ISI-1 and 2-RI-ISI-1.

4.3 Licensee's Proposed Alternative and Basis

The SQN Units 1 and 2 RI-ISI program was developed in accordance with the provisions of WCAP-14572, Revision 1-NP-A; WCAP-14572, Revision1-NP-A, Supplement 1; and WCAP-14572, Supplement 2 and is contained in RRs 1-RI-ISI-1 and 2-RI-ISI-1.

The configuration and the size of branch connections that are NPS 2 and smaller and socket welds prohibit the performance of volumetric examinations that would provide any meaningful results. The licensee asserts that the performance of a visual examination (VT-2 examination) during a system pressure test provides reasonable assurance of continued structural integrity.

The steam generator blowdown system has piping segments that contain inaccessible butt welds. The other welds in these segments are socket welds. The licensee requests to perform VT-2 examination during a system pressure test on the entire segment each outage at each unit. The licensee asserts that the performance of the VT-2 examination during a system pressure test provides reasonable assurance of continued structural integrity for the welds in the steam generator blowdown system piping segments that contain socket welds and butt welds that are inaccessible.

The containment spray system piping segments that contain socket welded piping cannot be isolated from the containment spray headers; therefore, the segments cannot be pressurized for a VT-2 examination. Any attempt to pressurize these segments would result in spraying down the upper containment with borated water. The licensee proposes to volumetrically examine selected butt welds in these segments to provide reasonable assurance of continued structural integrity.

4.4 NRC Staff Evaluation

WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," was approved for use on December 15, 1999. On June 22, 2006, WCAP-14572, Revision 1-NP-A, Supplement 2 was approved. The supplement made changes to the inspection requirements for socket welds and branch connections of NPS 2 and smaller. These changes are the same as the licensee's request to perform VT-2 examinations during a pressure test for socket welds and branch connections of 2 NPS and smaller every refueling outage when there is no external degradation mechanism. Based on the staff's approval of WCAP-14572, Revision 1-NP-A, Supplement 2, the staff finds this portion of the licensee's request acceptable.

The second part of the licensee's relief request proposes to perform VT-2 examinations during a pressure test for two pipe segments in the steam generator blowdown system that contain a 2-inch butt weld located in a containment penetration sleeve. The remaining welds in these segments are socket welds. The butt weld associated with each of these segments is located inside a containment penetration sleeve. To require the licensee to volumetrically examine these 2-inch butt welds would be a significant burden, because the licensee would need to redesign the piping and the containment penetration. The staff finds the licensee's proposal to perform a VT-2 examination during a pressure test for the subject pipe segments in the steam generator blowdown system every refueling outage should provide reasonable assurance of structural integrity. Typically, the VT-2 inspections are only performed once per period. The licensee's proposal performs the examinations more frequently. Therefore, the staff finds that to require volumetric examination of the 2-inch butt welds would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Two containment spray system segments on each unit contain socket welded piping that cannot be isolated from the containment spray headers. Therefore, the segments cannot be pressurized for a VT-2 examination. Any attempt to pressurize these segments would result in spraying down the upper containment with borated water. This would then require significant cleanup and would, therefore, create a significant burden for the licensee. The licensee will continue to volumetrically inspect selected butt welds in these segments. The staff finds that the volumetric and visual examinations should provide adequate assurance that any patterns of degradation will be identified and corrected for the subject containment spray segments, and that there is reasonable assurance of structural integrity of the two segments in the containment spray system at each unit. Therefore, the staff finds that the requirement to pressurize the segments for a VT-2 examination would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

5.0 CONCLUSIONS

Based on the information provided in the licensee's submittals and the above discussion, the NRC staff has determined that the alternative proposed in RRs 1-RI-ISI-1 and 2-RI-ISI-1 provides an acceptable level of quality and safety and, therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative is authorized for the third 10-year ISI inspection intervals at SQN Units 1 and 2, which began June 1, 2006.

For RRs 1-RI-ISI-2 and 2-RI-ISI-2, the NRC staff concluded that complying with the Code requirements as modified by the implementation of a RI-ISI program would result in hardship or difficulty without a compensating increase in the level of quality and safety and the proposed alternative provides reasonable assurance of structural integrity of the affected components. Therefore, RRs 1-RI-ISI-2 and 2-RI-ISI-2 are authorized, pursuant to 10 CFR 50.55a(a)(3)(ii), for the third 10-year ISI intervals at SQN Units 1 and 2, which began June 1, 2006.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this or previous relief requests remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

6.0 REFERENCES

1. Letter from P. L. Pace, TVA to NRC, "Sequoyah Nuclear Plant - (SQN) Risk-Informed Inservice Inspection (RI-ISI) Program and Updated Inservice Inspection (ISI) Program For Third 10-Year Interval," dated April 21, 2006.

2. Letter from G. Morris, TVA to NRC, "Sequoyah Nuclear Plant (SQN) Units 1 and 2 - Response to Request for Additional Information (RAI) for Risk-Informed Inservice Inspection (RI-ISI) Program Update," dated April 16, 2007.

3. Letter from P. Salas, TVA to NRC, “Sequoyah Nuclear Plant - Request for Approval of the SQN American Society of Mechanical Engineers (ASME) Section XI Alternative Inservice Inspection Program - Risk Informed Inservice Inspection (RI-ISI),” dated March 23, 2001.

4. Letter from NRC to TVA, “Sequoyah Nuclear Plant, Units 1 and 2 - Request for Additional Information on a Proposed Risk-Informed Inservice Inspection Program (TAC Nos. MB1566 and MB1567),” dated July 13, 2001.

5. Letter from P. Salas, TVA to NRC, “Sequoyah Nuclear Plant (SQN) - Response to Request for Additional Information (RAI) Regarding Risk-Informed Inservice Inspection (RI-ISI) Program,” dated August 31, 2001.

6. Letter from P. Salas, TVA to NRC, “Sequoyah Nuclear Plant (SQN) - Additional Information (RAI) Regarding Risk-Informed Inservice Inspection (RI-ISI Program),” dated October 3, 2001.

7. Letter from NRC to TVA, “Sequoyah Nuclear Plant, Units 1 and 2 - Approval of Relief Requests Regarding Risk-Informed Inservice Inspection Program (TAC Nos. MB1566 and MB1567),” dated October 19, 2001.

8. Letter from NRC to TVA, “Sequoyah Nuclear Plant, Units 1 and 2 - Clarification of Relief Requests Regarding Risk-Informed Inservice Inspection Program (TAC Nos. MB1566 and MB1567),” dated October 23, 2001.

9. WCAP-14572, Revision 1-NP-A, “Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report,” dated February 1999.

10. Letter from T. H. Essig, NRC, to Mr. Lou Liberatori, Westinghouse Owners Group, Safety Evaluation of Topical Report WCAP-14572, Revision 1, “Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report,” dated December 15, 1998.

11. WCAP-14572, Supplement 2, “Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications,” dated May 2004.

12. Letter from NRC, dated May 1, 2006, Final Safety Evaluation for Pressurized Water Reactor (PWR) Owners Group Topical Report WCAP-14572, REVISION 1-NP-A, Supplement 2, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications (TAC No. MC3979),” dated May 1, 2006.

13. Letter from F. P. "Ted" Schiffley, II, PWR Owners Group, PWR Owners Group, “Transmittal of NRC-Approved Topical Report WCAP-14572 Revision 1-NP-A,. Supplement 2 Revision 1-NP-A “Pressurized Water Reactor Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications,” September 2006 (TAC NO. MC3979), PAMSC-0076, Revision 2,” dated October 2, 2006.

14. NRC Regulatory Guide 1.174, Revision 1, An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis, dated November 2002.

15. NRC Regulatory Guide 1.178, Revision 1, *An Approach for Plant-Specific Risk-Informed Decision Making: Inservice Inspection of Piping*, dated September 2003.

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17. NEI 04-05, "Living Program Guidance to Maintain Risk-Informed Inservice Inspection Programs for Nuclear Plant Piping Systems," dated April 2004.

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