October 19, 2001

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - CLARIFICATION OF RELIEF REQUESTS REGARDING RISK-INFORMED INSERVICE INSPECTION PROGRAM (TAC NOS. MB1566 AND MB1567)

Dear Mr. Scalice:

In a letter dated March 23, 2001, as supplemented on August 31 and October 3, 2001, the Tennessee Valley Authority (TVA) requested approval of a risk-informed inservice inspection (RI-ISI) program for Class 1 and Class 2 piping welds as an alternative to the current ISI program at the Sequoyah Nuclear Plant, Units 1 and 2. The proposed RI-ISI program was developed in accordance with Westinghouse Owners Group Topical Report WCAP-14572, Revision 1-NP-A. The results of our review indicate that the proposed RI-ISI program is an acceptable alternative to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for inservice inspection of Code Class 1 piping, Categories B-F and B-J welds and Class 2 piping, Categories C-F-1 and C-F-2 welds. Therefore, TVA's request for relief is authorized pursuant to the regulatory requirements in 10 CFR 50.55a(a)(3)(i) for the second 10-year ISI interval on the basis that the alternative provides an acceptable level of quality and safety.

In addition, TVA submitted Relief Requests 1-RI-ISI-2 and 2-RI-ISI-2 for Sequoyah, Units 1 and 2, which requested performing visual VT-2 examinations each refueling outage as an alternative to the volumetric examinations specified in Code Case N-577 and in WCAP-14572, Revision 1-NP-A, for those high safety significant ASME Code Class 1 socket welds identified in the RI-ISI program. The U.S. Nuclear Regulatory Commission (NRC) staff concurs that volumetric examination of socket welds is inconclusive and impractical due to the geometric limitations imposed by a socket weld. The staff also concurs that it is not necessary to perform the Code-required surface examination of socket welds in the absence of an environment which would cause outside surface-initiated flaws. Therefore, TVA's proposed alternative examination method is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) on the basis that performing either the volumetric or the surface examinations of these socket welds would result in unusual difficulty without a compensatory increase in the level of quality and safety.

The enclosed Safety Evaluation authorizes application of the proposed RI-ISI program and Relief Requests 1-RI-ISI-2 and 2-RI-ISI-2 during the second 10-year ISI interval and will be implemented during the second inspection periods of the current inspection interval of Sequoyah Nuclear Plant, Units 1 and 2.

If you have questions or comments on the enclosed Safety Evaluation, please contact Mr. Ronald W. Hernan, the NRC Project Manager assigned to Sequoyah.

Sincerely,

/RA by Herbert N. Berkow for/
Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosure: Safety Evaluation

cc w/enclosures: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RISK-INFORMED INSERVICE INSPECTION PROGRAM

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

TENNESSEE VALLEY AUTHORITY

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

In a submittal to the U.S. Nuclear Regulatory Commission (NRC or the Commission) dated March 23, 2001, as supplemented on August 31, 2001, and October 3, 2001 (Refs. 1, 2, and 3), the Tennessee Valley Authority (the licensee) proposed a risk-informed inservice inspection (RI-ISI) program as an alternative to a portion of their current ISI program for Sequoyah Nuclear Plant, Units 1 and 2. The scope of the RI-ISI program is limited to the American Society of Mechanical Engineers (ASME) Code Class 1 and Class 2 piping. The licensee's RI-SI program was developed in accordance with the methodology contained in the Westinghouse Owners' Group (WOG) Topical Report, WCAP-14572, Revision 1-NP-A (Ref. 4), which was previously reviewed and approved by the NRC staff. The RI-ISI program proposed by the licensee was reviewed pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i). In addition, the licensee submitted Relief Requests 1-RI-ISI-2 and 2-RI-ISI-2 for Sequoyah, Units 1 and 2, which requested performance of visual VT-2 examinations each refueling outage as an alternative to the volumetric examinations specified in Code Case N-577 and in WCAP-14572, Rev. 1-NP-A, for those high safety significant (HSS) ASME Code Class 1 socket welds identified in the RI-ISI program. This relief request was reviewed by the NRC staff pursuant to 10 CFR 50.55a(a)(3)(ii).

2.0 BACKGROUND

2.1 Applicable Requirements

Per 10 CFR 50.55a(g), ISI of the ASME Code Class 1, 2, and 3 components must be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components" (hereinafter called Code) and applicable addenda, except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). As stated, in part, in 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that the proposed alternatives would provide an acceptable level of quality and safety or if the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

ENCLOSURE

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements set forth in the Code, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that ISI of components conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

For Sequoyah Units 1 and 2, the applicable editions of the Code for the second 10-year ISI interval are the 1989 Edition and the 1995 Edition through 1996 Addenda of ASME Section XI, respectively. Both units of Sequoyah are currently in the second period of the second 10-year ISI interval with a October 2001 outage for Unit 1 and a March 2002 outage for Unit 2 in the current inspection period prior to their entry into the third period of the second 10-year interval on December 16, 2002.

2.2 Summary of Proposed Approach

In the licensee's proposed RI-ISI program, piping failure potential estimates were determined using a software program contained in Supplement 1 to Reference 4, entitled "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," which utilizes probabilistic fracture mechanics technology, industry piping failure history, plant-specific piping failure history, and other relevant information. Using the failure potential and supporting insights on piping failure consequences from the licensee's probabilistic risk assessment (PRA), safety significance ranking of piping segments was established to determine inspection locations. The program maintains the fundamental requirements of the Code, such as the examination technique, frequency, and acceptance criteria. The RI-ISI program is intended to reduce the number of required examination locations significantly while maintaining an acceptable level of quality and safety.

The licensee plans to implement the RI-ISI program by performing the examinations required under the program during the planned outages of the second inspection period of the second 10-year ISI interval. Other non-related portions of the Code requirements, as well as the ongoing augmented inspection programs at both units of Sequoyah, will remain unchanged. The RI-ISI program follows a previously approved methodology delineated in Reference 4.

3.0 EVALUATION

Pursuant to 10 CFR 50.55a(a)(3), the NRC staff has reviewed and evaluated the licensee's proposed RI-ISI program, including those portions related to the applicable methodology and processes contained in Ref. 3, based on guidance and acceptance criteria provided in Regulatory Guides (RGs) 1.174 (Ref. 5) and 1.178 (Ref. 6) and in Standard Review Plan (SRP) Chapter 3.9.8 (Ref. 7).

3.1 Proposed Changes to the ISI Program

The scope of the licensee's proposed RI-ISI program is limited to ASME Class 1 and Class 2 piping only. The RI-ISI program was proposed as an alternative to the existing ISI program, which is based on the requirements of the Code. A general description of the proposed changes to the ISI program was provided in Sections 3 and 5 of the licensee's submittal (Ref. 1).

The NRC staff reviewed the proposed RI-ISI program against the guidelines contained in previously approved WCAP-14572, Rev. 1-NP-A, which states, in part, that the SRRA computer models are to be used to estimate the failure probabilities of the structural elements in each of the piping segments. In Reference 2, the licensee states that the failure probabilities for Sequoyah piping segments were all derived using the SRRA software program. This is consistent with the guidelines in previously approved WCAP-14572, Rev. 1-NP-A. Thus, the staff concludes that the licensee's application of WCAP-14572, Rev. 1-NP-A approach at Sequoyah is an acceptable alternative to the current piping ISI requirements with regard to the number, locations, and methods of inspections and, hence, provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(a)(3)(i).

In Table 5-1 and 5-2 of Reference 2, a comparison of inspection location selection between the current ISI program and the proposed RI-ISI program is provided. The staff finds that the information submitted adequately defines the proposed changes to the RI-ISI program.

3.2 Engineering Analysis

In accordance with the guidance provided in RGs 1.174 and 1.178 (Refs. 5 and 6), the licensee provided the results of an engineering analysis of the proposed changes, using a combination of traditional engineering analysis and PRA. The licensee stated that the results of the engineering analysis demonstrate that the proposed changes are consistent with the principle of defense-in-depth. This is accomplished by evaluating a location's susceptibility to each potential degradation mechanism that may be a precursor to leak or rupture and then performing an independent assessment of the consequence of a failure at that location. No changes to the evaluation of design basis accidents in the final safety analysis report are being made by the RI-ISI process. Therefore sufficient safety margins will be maintained.

The licensee stated that the applicable aspects of the ASME Code not affected by the proposed alternative RI-ISI program and the ongoing augmented inspection programs will be retained. This is consistent with approved WCAP-14572, Rev. 1-NP-A; therefore, it is acceptable.

TVA submitted Relief Requests 1-RI-ISI-2 and 2-RI-ISI-2 for Sequoyah, Units 1 and 2, which requested performing visual VT-2 examinations during each refueling outage as an alternative to the volumetric examinations specified in Code Case N-577 and in WCAP-14572, Rev. 1-NP-A, for those HSS ASME Code Class 1 socket welds identified in the RI-ISI program. The licensee indicated that Code Case N-577 has been revised to allow the substitution of the VT-2 examination method for all damage mechanisms on socket welds selected as HSS. This request is reasonable because the volumetric examination is inconclusive and impractical due to the geometric limitations imposed by a socket weld. However, the staff notes that Table IWB-2500-1 of the Code requires surface examination, not volumetric examination, at the

socket welds, and surface examination (i.e., liquid penetration examination) is an effective method for discovery of potential surface flaws on the outside surface, and specifically, flaws induced by low-cycle, high-bending stress thermal fatigue or by external chloride stress corrosion cracking (ECSCC). The licensee indicated that the Code Class 1 socket weld piping is not located in areas that are subject to an environment promoting ECSCC, and an outside surface-initiated flaw has a very low probability of occurrence due to the inclusion of thermal cyclic loads in the piping design. Therefore, these conditions do not exist in the Code Class 1 piping at Sequoyah. As for a potential outside surface flaw caused by vibration-induced fatigue, such a flaw is likely to take a long period for initiation. After the initiation phase, the flaw will likely propagate rapidly and cause the pipe to leak. Hence, the staff concludes that performance of a VT-2 visual examination is sufficiently effective and, therefore, acceptable. The staff notes that the revised Code Case N-577 has neither been issued nor been reviewed and approved by the NRC. Thus, the approval of this request is based on the technical soundness of applying VT-2 visual examination to specific conditions at Sequoyah, and should not be considered as an endorsement of the Code case. Pursuant to 10 CFR 50.55a(a)(3)(ii), the staff concurs that performing volumetric or surface examinations of these socket welds would result in unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the request to conduct VT-2 examinations as an alternative in each refueling outage for Category B-J socket welds is acceptable.

Piping systems within the scope of the proposed RI-ISI program were divided into piping segments. A pipe segment is defined as a portion of pipe length whose failure at any location within the segment will lead to the same consequence. Pipe segments are separated by flow splits and locations of pipe size changes, and include piping to a point at which a pipe break could be isolated. The licensee reported no deviations from the identification and definition of segments in WCAP-14572, Rev. 1-NP-A, and their process is, therefore, acceptable.

Piping failure mechanisms identified by the licensee include fatigue, stress corrosion cracking, thermal striping/stratification, erosion/corrosion/wastage, and vibratory fatigue. The failure probabilities for the Sequoyah piping segments were all derived using the Westinghouse SRRA software program. This is consistent with the guidelines in WCAP-14572, Rev. 1-NP-A, and in conformance with SRP 3.9.8. The licensee reported a deviation in the WCAP-14572, Rev. 1-NP-A methodology regarding credit taken for leak detection when calculating pipe failure probabilities. WCAP-14572, Rev. 1-NP-A allows credit for detecting (and isolating, repairing, or otherwise terminating a potential accident sequence) a leak in the reactor coolant system (RCS) piping before it develops into a pipe break for piping inside of containment. This credit reflects the highly developed leak detection systems used to monitor leakage from the reactor coolant piping (RCP). In Reference 2, the licensee further states that detection of a leak before break is plausible to any non-RCS segment located inside the containment that interfaces with the RCS by use of radiation and sump level monitors that can detect a leak in the segment as reliably as that of an RCS leak. Since the segments are subject to essentially the same leak detection capabilities as that of an RCS leak, the extension of credit for leak detection in these segments, is reasonable and acceptable. The licensee has developed the consequence of each segment break based on the direct and indirect effects of the segment failure. The licensee has reported no deviations from the consequence characterization methodology in WCAP-14572, Rev. 1-NP-A, and, therefore, their analyses are acceptable.

3.3 Probabilistic Risk Assessment

The licensee has used Revision 1 of its PRA model to support the RI-ISI submittal. The current estimates of core damage frequency (CDF), *Sequoyah Nuclear Plant Unit 1 Probabilistic Risk Assessment Individual Plant Examination, Revision 1, RIMS No. B38 960806 800* and large early release frequency (LERF), *LERF Models for Sequoyah and Watts Bar, January 2000, RIMS No. B45 000516 001,* are 4.0E-5/yr and 8.6E-7/yr, respectively. This revision was reviewed with the WOG PRA Peer Review Certification process and a draft revision 2 was developed. A qualitative assessment of the effects of the findings and observations of this peer review of the draft Revision 2 based on the results of Revision 1 of the PRA, was performed. The licensee stated that based on the conclusion of their assessment, Revision 1 of the PRA was fully adequate for use to support the RI-ISI program.

WCAP-14572, Rev. 1-NP-A, requires that functions relied upon to mitigate external events, including fires and operational transients that are beyond the scope of the PRA, be systematically included in the categorization by the engineering team members and the expert panel. The licensee did not report a deviation in this area and, therefore, the use of a PRA derived from an internal event, full-power operation Individual Plant Examination (IPE) is acceptable.

By letter dated December 9, 1994, the licensee indicated that the two units are essentially identical and the results are applicable to both Units 1 and 2. The staff evaluation report, dated May 15, 1995, concluded that the Sequoyah IPE satisfied the intent of Generic Letter 88-20. No weaknesses or concerns were identified regarding the models or the parameters used in the IPE. The licensee stated that administrative procedures provide guidance for the periodic update of the PRA. The continuous use and maintenance of the PRA provides further opportunities to identify inaccuracies and inappropriate assumptions, if any, in the PRA models.

The staff did not review the PRA analysis to assess the accuracy of the quantitative estimates. Quantitative results of the PRA are used, in combination with a quantitative characterization of the pipe segment failure likelihood, to support the assignment of segments into broad safety significance categories reflecting the relative importance of pipe segment failures on CDF and LERF. Inaccuracies in the models or assumptions large enough to invalidate the broad categorizations developed to support the RI-ISI should have been identified in the licensee's or in the staff's review. Minor errors or inappropriate assumptions will only affect the consequence categorization of a few segments and will not invalidate the general results or conclusions. The staff finds that the quality of the PRA is sufficient to support the submittal.

The licensee stated in Reference 2 that the risk ranking and change in risk calculations were performed according to the guidance provided in Section 4.4.2 of WCAP-14572, Rev. 1-NP-A, aside from the one deviation discussed in Section 3.2 of this Safety Evaluation. The submittal included estimates for Sequoyah Unit 1 on the change in CDF and LERF associated with replacing the current ASME Section XI weld inspection locations for Code Class 1and 2 piping with the proposed RI-ISI inspection locations. The change in CDF is estimated to be about -3.0E-6/yr with and -2.9E-6/yr without operator action. The change in LERF is estimated to be -8.0E-8/yr with and -8.0E-8/yr without operator action. For Unit 2, the change in CDF is estimated to be about -1.4E-6/yr with and -3.17E-5/yr without operator action. The change in LERF is estimated to be -4.4E-8/yr with and -5.0E-8/yr without operator action.

The licensee did not submit estimates for the other risk change criteria in Section 4.4.2 of WCAP-14572, Rev. 1-NP-A, but stated in Reference 2 that all the change in risk calculations were performed according to the guidance on page 213 of the WCAP (as applicable) and all four criteria for evaluating the results were applied. Adjustments were made to add segments and eventually inspection locations until all the four criteria discussed on pages 214 and 215 of the WCAP were met. Based on the use of the approved methodology and on the reported results, the staff finds that any change in risk associated with the implementation of the RI-ISI program is small and consistent with the intent of the Commission's Policy Statement (Ref. 8) and, therefore, is consistent with RG 1.178.

3.4 Integrated Decisionmaking

The proposed RI-ISI program presents an integrated approach that considers in concert the traditional engineering analysis, the risk evaluation, and the implementation and the performance monitoring of piping. This is consistent with the guidelines of RG 1.178.

The selection of pipe segments to be inspected is described in Section 3.8 of Reference 1 using the results of the risk category rankings and other operational considerations. Table 5-1 of Reference 1 provides a summary table comparing the number of inspections required under the existing ASME Section XI ISI program at Sequoyah Units 1 and 2 with the alternative RI-ISI program. The licensee stated that, in general, it used the methodology described in WCAP-14572, Rev. 1-NP-A to guide the selection of the number and the location of examination elements within the piping segments.

For some segments, the licensee deviated from the application of the statistical method to determine the number of location for inspection that is described in Section 3.7.2 in the WCAP. The WCAP states that the "probability of an unacceptable flaw in the segments 'most likely to fail' weld (or typical weld, if they are viewed as clones) at the current age of the weld," should be considered in the statistical method as the input parameter, "Probability of a Flaw." This input is combined with the number of welds and with the conditional probability of leak/year/weld to identify the number of inspections required to satisfy the statistical constraints. The licensee reported that, if the number of inspections required to satisfy the statistical criteria exceeds 10, the welds are re-evaluated and those welds "most likely to fail" are identified and removed from the population of welds under consideration. All these welds will be inspected. The statistical evaluation is continued by selecting the new "most likely to fail" weld from the remaining population and adjusting the input parameters to reflect the likelihood of this weld's failure and the reduced number of welds. This process is repeated until the final population (e.g., excluding all the welds removed from the population and that will be inspected) satisfies the statistical criteria with less than ten inspections. The total number of inspections includes all the welds removed from the population and the final number required to satisfy the statistical criteria.

The WCAP methodology defines segments as lengths of piping with identical consequences. Therefore, identifying and inspecting those welds with the highest likelihood of failure within a segment is identical to identifying and inspecting those welds with the highest safety significance. The statistical calculation is applied to determine the number of inspections required in the remaining population to satisfy the statistical criteria in the WCAP. This approach is consistent with the concept that, by focusing inspections on the most safety

significant welds, the number of inspections can be reduced while at the same time maintaining public health and safety and, therefore, this approach is acceptable.

In Reference 3 the licensee identified three segments in each unit where the selection of the number of locations for inspection deviated from the WCAP, and from the above method. The licensee determined that special circumstances unforeseen in the WCAP resulted in the statistical method not being applicable even though the characteristics of the segments and associated elements were within the parameters of applicability defined in the WCAP. These segments are composed of thin-walled piping and had no construction Code radiograph of the welds for final acceptance. However, surface examination was performed on each weld of the segments. The licensee stated that the criterion for use of the statistical method with existence of an unacceptable flaw determined in accordance with the ASME Code, Section XI, as the flaw depth to thickness ratio of 10% at the current age of the plant defeats the statistical method of determining the number of examination locations for these thin-walled piping. The staff has taken into consideration the material type, the fracture toughness, pressure/temperature rating, results of previous inservice examinations, and existing degradation mechanism to evaluate licensee's examination locations. The staff accepts the licensee's rationale in regard to the criteria for selection of the number of examination locations for these segments.

The staff finds that the licensee's selection process uses "defense-in-depth" considerations and is generally consistent with the WCAP-14572, Rev. 1-NP-A. The staff has evaluated the deviations from the WCAP topical report identified by the licensee and finds them acceptable.

WCAP-14572, Rev. 1-NP-A describes targeted examination volumes (typically associated with welds) and methods of examination based on the type(s) of degradation expected. The staff has reviewed these guidelines and has determined that, if implemented as described, the RI-ISI examinations should result in improved discovery of service-related discontinuities over that currently provided by the Code.

The objective of ISI required by the Code is to identify conditions (i.e., flaw indications) that are precursors to leaks and ruptures in the pressure boundary that may impact plant safety. Therefore, the RI-ISI program must meet this objective to be found acceptable for use. Further, since the RI program is based on inspection for cause, element selection should target specific degradation mechanisms.

Section 4 of WCAP-14572, Rev. 1-NP-A provides guidelines for the areas and/or volumes to be inspected as well as the examination method, acceptance standard, and evaluation standard for each degradation mechanism. Based on a review of the cited portion of WCAP-14572, Rev. 1-NP-A, the staff concludes that the examination methods are appropriate since they are selected based on specific degradation mechanisms, pipe sizes, and materials of concern. The licensee reported no deviations in this area from the WCAP-14572, Rev. 1-NP-A methodology and, therefore, its evaluation is acceptable.

3.5 Implementation and Monitoring

Implementation and performance monitoring strategies require careful consideration by the licensee and are addressed in Element 3 of RG 1.178 and SRP 3.9.8. The objective of Element 3 is to assess performance of the affected piping systems under the proposed RI-ISI program by implementing monitoring strategies that confirm the assumptions and analyses

used in the development of the RI-ISI program. To approve an alternative pursuant to 10 CFR 50.55a(a)(3)(i), implementation of the RI-ISI program, including inspection scope, examination methods, and methods of evaluation of examination results, must provide an adequate level of quality and safety.

In Reference 1, the licensee stated that upon approval of the RI-ISI program, procedures that comply with the WCAP-14572, Rev. 1-NP-A guidelines will be prepared to implement and monitor the RI-ISI program. The licensee confirmed that the applicable portions of the Code not affected by the change, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements would be retained.

The licensee stated in Section 4 of Reference 1 that the RI-ISI program is a living program and its implementation will require feedback of new relevant information to ensure the appropriate identification of HSS piping locations. Reference 1 also stated that as a minimum, risk ranking of piping segments will be reviewed and evaluated every ISI period and that significant changes may require more frequent adjustments as directed by any NRC Bulletin or Generic Letter or by industry and plant-specific feedback.

The proposed periodic reporting requirements meet existing ASME Code requirements and applicable regulations and, therefore, are considered acceptable. The staff finds that the proposed process for RI-ISI program updates meets the guidelines of RG 1.174 that risk-informed applications should include performance monitoring and feedback provisions; therefore, the process for program updates is acceptable.

4.0 CONCLUSION

As stated in 10 CFR 50.55a(a)(3)(i), alternatives to regulatory requirements are permitted when authorized by the NRC if the applicant demonstrates that the alternative provides an acceptable level of quality and safety. In this case, the licensee's proposed alternative is to use the RI process described in the NRC-approved report WCAP-14572, Rev. 1-NP-A. As discussed in Section 3.0 above, the staff concludes that the licensee’s proposed RI-ISI program, as described in the submittal, will provide an acceptable level of quality and safety with regard to the number of inspections, locations of inspections, and methods of inspection.

The staff finds that the results of different elements of the engineering analysis are considered in an integrated decision-making process. The impact of the proposed changes in the ISI program is founded on the adequacy of the engineering analysis and acceptable estimation of changes in plant risk in accordance with RG 1.174 and RG 1.178 guidelines.

The Sequoyah methodology also considers implementation and performance monitoring strategies. Inspection strategies ensure that failure mechanisms of concern have been addressed and there is adequate assurance of detecting damage before structural integrity is affected. The risk significance of piping segments is taken into account in defining the inspection scope for the RI-ISI program.

System pressure tests and visual examination of piping structural elements will continue to be performed on all Code Class 1and 2 systems in accordance with the ASME Code Section XI

program. The RI-ISI program applies the same performance measurement strategies as existing ASME Code requirements.

The Sequoyah risk-informed methodology provides for conducting an analysis of the proposed changes using a combination of engineering analysis with supporting insights from a PRA. Defense-in-depth and quality are not degraded in that the methodology provides reasonable confidence that any reduction in existing inspections will not lead to degraded piping performance when compared to existing performance levels. Inspections are focused on locations with active degradation mechanisms as well as selected locations that monitor the performance of system piping.

There were three deviations from the WCAP in the submittal; the application of leak-before-break in non-RCS piping, the modified statistical methodology, and the application of an alternative method to select the number of locations for inspection in three segments in each unit. The staff has reviewed these deviations as described by the licensee and finds them acceptable in this application.

In addition, the licensee submitted Relief Requests 1-RI-ISI-2 and 2-RI-ISI-2 for Sequoyah, Units 1 and 2, which requested performance of visual VT-2 examinations each refueling outage as an alternative to the volumetric examinations specified in Code Case N-577 and in WCAP-14572, Rev. 1-NP-A, for those HSS ASME Code Class 1 socket welds identified in the RI-ISI program. The staff concurs that volumetric examination of socket welds is inconclusive and impractical due to the geometric limitations imposed by a socket weld. The staff also concurs that to perform Code-required surface examination of socket welds is not useful due to the absence of an environment that would cause outside surface-initiated flaws. Therefore, the licensee's proposed alternative examination method is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) on the basis that performing either volumetric or surface examinations of these socket welds would result in unusual difficulty without a compensating increase in the level of quality and safety, and that the licensee's proposed alternative provides reasonable assurance of structural integrity.

As discussed above, the staff's review of the licensee's proposed RI-ISI program concludes that the program is an acceptable alternative to the current ISI program, which is based on ASME Code, Section XI, requirements for Code Class 1, Categories B-F and B-J welds and for Code Class 2, Categories C-F-1 and C-F-2 welds. Therefore, the licensee's proposed RI-ISI program is authorized for the second 10-year ISI interval pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the request provides an acceptable level of quality and safety. In addition, Relief Requests 1-RI-ISI-2 and 2-RI-ISI-2 are authorized for the second 10-year ISI interval pursuant to 10 CFR 50.55(a)(3)(ii) on the basis that performing the examinations would result in unusual difficulty without a compensating increase in the level of quality and safety.

5.0 REFERENCES

1. Letter, dated March 23, 2001, P. Salas (Tennessee Valley Authority, Licensing and Industry Affairs Manager) to U.S. Nuclear Regulatory Commission, containing *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)*.

2. Letter, dated August 31, 2001, P. Salas (Tennessee Valley Authority, Licensing and Industry Affairs Manager) to U.S. Nuclear Regulatory Commission, containing *Sequoyah Nuclear Plant (SQN) - Response to Request for Additional Information (RAI) Regarding Risk0Informed Inservice Inspection (RI-ISI) Program.*

3. Letter, dated October 3, 2001, P. Salas (Tennessee Valley Authority, Licensing and Industry Affairs Manager) to U.S. Nuclear Regulatory Commission, containing *Sequoyah Nuclear Plant (SQN) - Additional Information (RAI) Regarding Risk-Informed Inservice Inspection (RI-ISI Program)*.

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

5. NRC Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, July 1998.

6. NRC Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decision Making: Inservice Inspection of Piping*, September 1998.

7. NRC NUREG-0800, Chapter 3.9.8, *Standard Review Plan for Trial Use for the Review of Risk-Informed Inservice Inspection of Piping*, May 1998.

8. USNRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," *Federal Register*, Vol. 60, p. 42622, August 16, 1995.

Principal Contributors: S. Dinsmore, NRR
P. Patnaik, NRR

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Mr. J. A. Scalice
Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

cc:
Mr. Karl W. Singer, Senior Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Jon R. Rupert, Vice President (Acting)
Engineering & Technical Services
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Richard T. Purcell
Site Vice President
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

General Counsel
Tennessee Valley Authority
ET 11A
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. Robert J. Adney, General Manager
Nuclear Assurance
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Mark J. Burzynski, Manager
Nuclear Licensing
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Pedro Salas, ManagerLicensing and
Industry Affairs
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

Mr. D. L. Koehl, Plant Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

Mr. Russell A. Gibbs
Senior Resident Inspector
Sequoyah Nuclear Plant
U.S. Nuclear Regulatory Commission
2600 Igou Ferry Road
Soddy Daisy, TN 37379

Mr. Lawrence E. Nanney, Director
Division of Radiological Health
Dept. of Environment & Conservation
Third Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

County Executive
Hamilton County Courthouse
Chattanooga, TN 37402-2801

Ms. Ann P. Harris
341 Swing Loop Road
Rockwood, Tennessee 37854