May 4, 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: SAFETY EVALUATION OF SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2, REQUEST FOR RELIEF RI-IST-1, RISK-INFORMED INSERVICE TESTING PROGRAM FOR SELECTED VALVES (TAC NOS. MA9097 AND MA9098)

Dear Mr. Scalice:

By letter dated April 27, 2000, the Tennessee Valley Authority (TVA), submitted a request for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI requirements under Title 10, *Code of Federal Regulations*, Part 50, Section 55a(a)(3)(i) for the second 10-year Inservice Testing (IST) Program at the Sequoyah Nuclear Plant, Units 1 and 2. The proposed request for relief was to implement a limited scope Risk-Informed IST (RI-IST) Program for selected valves.

The U.S. Nuclear Regulatory Commission (NRC) staff has completed its review of this request. Although TVA provided some quantitative risk information in its submittal, the information provided is not complete and does not address some important elements of an acceptable RI-IST Program, as presented in NRC Regulatory Guide (RG) 1.175. To support an adequate risk-informed submittal, TVA would be required to (1) provide additional information, such as the determination of the safety significance of the selected valves, (2) perform additional quantitative risk calculations, such as the aggregate risk resulting from implementing all of the proposed RI-IST Program extended test intervals, and (3) address traditional engineering considerations, such as defense-in-depth and safety margins. After discussions with the NRC regarding these additional information requirements, TVA elected to withdraw the request and notified the NRC of this decision by letter dated March 30, 2001.

Enclosed is our evaluation of the information provided by TVA for the proposed limited scope RI-IST Program. Based on this evaluation, the NRC staff finds that TVA's submittal is incomplete and, therefore, it is not possible to develop a defensible conclusion, based upon the guidance in RG 1.174 and RG 1.175, regarding the acceptability of the licensee's proposed limited scope RI-IST Program. However, these findings do not preclude the possibility of an acceptable non-risk-informed basis to support the request.

Sincerely,

/RA/
Ronald W. Hernan, Sr. Project Manager, 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/enclosure: See next page

May 4, 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: SAFETY EVALUATION OF SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2, REQUEST FOR RELIEF RI-IST-1, RISK-INFORMED INSERVICE TESTING PROGRAM FOR SELECTED VALVES (TAC NOS. MA9097 AND MA9098)

Dear Mr. Scalice:

By letter dated April 27, 2000, the Tennessee Valley Authority (TVA), submitted a request for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI requirements under Title 10, *Code of Federal Regulations*, Part 50, Section 55a(a)(3)(i) for the second 10-year Inservice Testing (IST) Program at the Sequoyah Nuclear Plant, Units 1 and 2. The proposed request for relief was to implement a limited scope Risk-Informed IST (RI-IST) Program for selected valves.

The U.S. Nuclear Regulatory Commission (NRC) staff has completed its review of this request. Although TVA provided some quantitative risk information in its submittal, the information provided is not complete and does not address some important elements of an acceptable RI-IST Program, as presented in NRC Regulatory Guide (RG) 1.175. To support an adequate risk-informed submittal, TVA would be required to (1) provide additional information, such as the determination of the safety significance of the selected valves, (2) perform additional quantitative risk calculations, such as the aggregate risk resulting from implementing all of the proposed RI-IST Program extended test intervals, and (3) address traditional engineering considerations, such as defense-in-depth and safety margins. After discussions with the NRC regarding these additional information requirements, TVA elected to withdraw the request and notified the NRC of this decision by letter dated March 30, 2001.

Enclosed is our evaluation of the information provided by TVA for the proposed limited scope RI-IST Program. Based on this evaluation, the NRC staff finds that TVA's submittal is incomplete and, therefore, it is not possible to develop a defensible conclusion, based upon the guidance in RG 1.174 and RG 1.175, regarding the acceptability of the licensee's proposed limited scope RI-IST Program. However, these findings do not preclude the possibility of an acceptable non-risk-informed basis to support the request.

Sincerely,
/RA/
Ronald W. Hernan, Sr. Project Manager, 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/enclosure: See next page

ADAMS ACCESSION NUMBER: ML011240271

OFFICE	PDII-2/PM		PDII-2/LA	E	PDII-2/SC	
NAME	RHernan		BClayton		RCorreia	
DATE	5/04/01		5/03/01		5/04/01	

OFFICIAL RECORD COPY

DISTRIBUTION:
PUBLIC (PUBLIC)
ACRS (RidsAcrsAcnwMailCenter)
PDII-2 Reading (Hardcopy)
SRosenberg (e-mail)
HBerkow (RidsNrrDlpmLpdii)
PFredrickson, RII (RidsRgn2MailCenter)
RCorreia (RidsNrrDlpmLpdii2)
OGC (RidsOgcRp)
BClayton (Hardcopy)
RHernan (Hardcopy)
TSullivan (RidsNrrDeEmcb)
LLund (RidsNrrDeEmcb)

SAFETY EVALUATION BY THE PROBABILISTIC SAFETY ASSESSMENT BRANCH

RISK-INFORMED INSERVICE TESTING PROGRAM FOR SELECTED VALVES

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1. INTRODUCTION

By letter dated April 27, 2000 (Ref. 1), the Tennessee Valley Authority (TVA), the licensee, submitted a request for relief from the nominal 3-month test frequency of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Operations & Maintenance Pump and Valve Inservice Test (IST) requirements for selected valves at the Sequoyah Nuclear Plant, Units 1 and 2 (SQN). The licensee states that the primary objective of the relief request is to reduce the challenges to routine/safe operation of the plant through reductions in the number of unnecessary periodic tests on certain valves while strengthening the effectiveness of the tests and overall plant safety. The licensee's relief request was submitted pursuant to Title 10, *Code of Federal Regulations*, Part 50, Section 55a(a)(3)(i), as a proposed alternative that would provide an acceptable level of quality and safety.

Subsequent to this submittal, the licensee submitted a separate relief request to specifically address the inservice testing of the eight containment spray (CS) valves (four at each unit) that were part of the April 27, 2000, submittal. This separate relief request was approved by the U.S. Nuclear Regulatory Commission (NRC) on September 20, 2000 (Ref. 2). Therefore, these CS valves are not addressed further as part of this evaluation of the licensee's April 27, 2000, relief request.

2. DESCRIPTION OF LICENSEE'S PROPOSED RISK-INFORMED IST (RI-IST) PROGRAM

The licensee used performance test data, coupled with risk-informed evaluations of the existing SQN Probabilistic Safety Assessment (PSA) and Maintenance Rule Program data, to establish new extended test frequencies for selected valves as an alternative to the quarterly test frequency specified in the ASME Code. Valves within the current SQN IST Program were identified for possible extension of their test frequencies and were evaluated in the following manner. The maintenance history, preventive maintenance programs, IST test data history, and trends of the IST Program valves were reviewed. Valves that exhibited low maintenance and low corrective action needs over the past 3 to 5 years, and that were consistently good test performers, were compiled into a list. The criterion for consistently good test performance was established as having no ASME Code test failures and/or associated corrective actions over the past 3 years. Valves that are currently tested only during cold shutdown or refueling were eliminated from the resulting list. Motor-operated valves (MOVs) were also eliminated from the list because there is already a separate initiative at SQN to develop the MOV Program in accordance with recent NRC Guidelines (i.e., NRC Generic Letter 96-05) and industry initiatives.

The licensee's process resulted in an initial list of 403 valves in 12 systems, which were then compared with the SQN PSA Program to identify the IST valves that are modeled in the SQN PSA Revision 1(PSA-1) model. Valves that were not modeled in the SQN PSA Program were eliminated unless they were similar in plant and/or unit-system configurations to the components that were modeled. This resulted in a list of 160 valves (80 at each unit) in 10 systems. This final valve population was then evaluated for the possible extension of test intervals as supported by the SQN risk-informed process. The selected valves were evaluated on a system or multi-system basis, but an evaluation of the aggregate impact resulting from extending the test interval for all the selected valves was not performed.

The licensee determined the impacts on core damage frequency (CDF) and large early release frequency (LERF) resulting from the decrease in the valves' test frequencies using the SQN PSA-1 model. The valves considered for this evaluation are currently tested on a quarterly basis (i.e., every 92 days). The test frequency considered in the proposed change assumed a test frequency of once every refueling cycle (i.e., 18 months). This increase in test interval equates to an increase of a factor of six in the time between periodic tests. It is stated by the licensee that its evaluation conservatively assumed that the failure rate of the valves, including common cause as well as independent failure rates, would increase linearly proportionally with the increase in test interval. However, it is also stated that in some cases an evaluation was developed to show that the increase in test interval would not affect the valve's failure rate. For these cases, the failure rates for the valves were not changed. The licensee did not identify for which valves an increased failure rate was used and for which valves an evaluation was performed to justify not increasing the failure rate.

The effect on LERF was evaluated through an integrated risk analysis of three of the ten systems (i.e., Purge Air [PA], Waste Disposal [WD], and Radiation Monitoring [RM]). The selected valves in these three systems are associated with the containment isolation function. However, the calculation of the impact on LERF associated with the other systems is not explicitly presented in the licensee's submittal. For these systems, the impacts on LERF were inferred, if possible, from the licensee's individual system evaluation discussions.

3. STAFF EVALUATION OF THE LICENSEE'S PROPOSED RI-IST PROGRAM

The licensee provided some information related to impacts on CDF and LERF as a result of the proposed extension in test intervals for selected valves. However, the licensee did not completely address some important elements of an acceptable RI-IST Program, as defined in Regulatory Guide (RG) 1.175 (Ref. 3). To support a complete risk-informed submittal, the licensee would need to provide additional information, such as the determination of the safety significance of the selected valves, perform additional quantitative risk calculations, such as the aggregate risk resulting from implementing all of the proposed RI-IST Program extended test intervals, and also address traditional engineering considerations, such as defense-in-depth and safety margins.

However, instead of providing additional information to support its relief request, the licensee decided to withdraw its risk-informed submittal Ref. 4). Due to this decision, the staff evaluated the information that was available from the licensee's initial risk-informed submittal to provide some insights into its acceptability and limitations. The staff used a simplified approach, based on RG 1.175, to evaluate the relief request and its potential impacts on CDF and LERF, which were then compared against the acceptance guidelines, consistent with the Commission's Safety Goal Policy Statement, as defined in RG 1.174 (Ref. 5).

3.1 Evaluation of the Validity of the SQN PSA and Its Application

To determine whether the SQN PSA used in support of the proposed limited scope RI-IST Program is of sufficient quality, scope, and detail, the staff evaluated the available information provided by the licensee in its submittal and considered the review findings on the original SQN individual plant examination (IPE). The staff's evaluation of the licensee's submittal focused on the capability of the licensee's PSA model to analyze the risks stemming from the proposed RI-IST Program and did not involve an in-depth review of the licensee's PSA.

The licensee indicated that it has continued to refine and improve the SQN PSA since the IPE was developed, such that it now includes external events, as well as internal events, recent design modifications, updated plant procedures, enhanced training programs, maintenance, and operational changes. The licensee also indicated that the current SQN PSA-1 model addresses the specific failure modes of valves (e.g., fails to open, fails to close, fails to remain open, fails to remain closed) and that some valves have more than one failure mode.

The licensee did not indicate if an industry/independent peer review on the SQN PSA-1 model had been conducted and did not address the revisions since its SQN IPE submittal that might be of importance to this application. The licensee did not provide any further insights into the quality, scope, and detail of its current PSA-1 model and did not provide any other information to support its suitability for this specific application.

The staff has identified a number of concerns regarding the SQN PSA-1 model as it applies to this specific application. First, the staff evaluation report on the SQN IPE noted that the licensee did not plan to maintain the IPE as a "living" analysis and document. The licensee did not provide a description of the quality control program for the SQN PSA, including the process for reviewing and accepting modifications to the models. Since the licensee has updated the analysis since the IPE submittal, it is not clear what controls the licensee used to ensure that the current model adequately reflects the current plant conditions, operations, and configurations.

Second, the SQN PSA-1 model CDF value is a factor of four less than the SQN IPE CDF, even though it is stated as including additional events (e.g., external events). There is no description of the major changes and assumptions that were incorporated into the revised models that might affect the components and systems that are the subject of this specific relief request. In particular, it is not clear if recovery of failed components is considered in the SQN PSA, which the RGs caution against, or if the operator actions and assumptions in the SQN PSA-1 are substantially different from those of the original IPE.

A number of other areas of concern have been identified, including:

- Valve safety significance is not determined (e.g., some of the selected valves may be of high safety significance but the proposed RI-IST Program would extend its testing intervals, which is cautioned against in RGs).
- Some specific system operations described in the licensee's submittal differ substantially from the original SQN IPE system operating descriptions and these differences are not addressed.
- Some of the described system valve group evaluations do not address all valves in the grouping.
- The conditions described to support the discovery of degraded performance do not address the potentially important failure modes for some valves.
- The valve selection process does not identify valves that may need improved or enhanced testing methods due to poor performance (i.e., the process only looked for good performers to increase test intervals).
- Important system and operator dependencies and assumptions that might impact this specific application are not addressed.
- Risks during shutdown operations are not addressed.
- Dual plant operations and configurations considering cross-unit dependencies are not addressed (e.g., Unit 1 operating at full power with Unit 2 in a refueling outage with vital, cross-unit systems and/or components out of service).
- Various plant/system operating configurations, other than those modeled by the SQN PSA, are not addressed (e.g., the PA system can be aligned such that one pair of lines - one supply and one exhaust line - can be open for up to 1000 hours per year during plant operating modes 1 through 4, which would provide a direct pathway outside containment that may be open at the time of the initiating event and may fail to close or fail to receive an isolation signal, resulting in a containment bypass).
- Uncertainties and sensitivities to key PSA modeling parameters and assumptions that might impact this specific application are not addressed.

The staff finds that the licensee has not completely addressed the issue of PSA quality for this specific application and, therefore, it is not possible to develop a defensible conclusion as to its acceptability.

3.2 Evaluation of the PSA Results and Insights

An acceptable element of risk-informed decisionmaking is to show that the proposed change meets several key principles. One of these principles is to show that the proposed change results in only a small increase in risk.

The SQN PSA-1 model establishes base CDF and LERF values of 3.8E-5/reactor-year and 4.45E-6/reactor-year, respectively. The licensee cited an integrated risk analysis and sensitivity studies, but these evaluations were not provided with the submittal; only the results were provided in the brief evaluation descriptions for each system. The change in CDF (ΔCDF) resulting from the proposed relief request was presented for individual system valve groups. The change in LERF (ΔLERF) resulting from the proposed relief request was only presented for a multi-system valve group consisting of the PA, WD, and RM systems. In addition, the ΔLERF values for control air (CA), safety injection (SI) and essential raw cooling water (ERCW) were inferred based on the individual system evaluation discussions, since explicit values were not presented. However, ΔLERF values were not presented and could not be inferred for steam generator blowdown (SGB) and the chemical and volume control system (CVCS). Based on the information provided in the submittal by the licensee, the following results were compiled.

SYSTEM	**ΔCDF ($rx\text{-}yr^{-1}$)**	**ΔLERF(IRA)[2] ($rx\text{-}yr^{-1}$)**	**ΔLERF(ALL)[3] ($rx\text{-}yr^{-1}$)**
Steam Generator Blowdown (SGB)[1]	3.0E-9		
Purge Air (PA)[2]	0.0E+0	1.1E-8	1.1E-8
Control Air (CA)[3]	2.0E-9		2.0E-9
Chemical and Volume Control (CVCS)[1]	5.4E-8		
Safety Injection (SI)[3]	1.2E-7		1.2E-7
Essential Raw Cooling Water (ERCW)[3]	2.5E-8		2.5E-8
Component Cooling (CC)	0.0E+0		0.0E+0
Waste Disposal (WD)[2]	0.0E+0	NA	NA
Radiation Monitoring (RM)[2]	0.0E+0	NA	NA
Total	**2.0E-7**	**1.1E-8**	**1.6E-7**

[1] No ΔLERF values were provided or could be inferred for SGB and CVCS.
[2] The ΔLERF(IRA) value is based on an integrated risk analysis that was performed for PA, WD, and RM that resulted in a single value for all three systems of 1.1E-8/reactor-year. This value is entered as the PA entry.
[3] The ΔLERF(ALL) values include the LERF(IRA) value, as noted in footnote 2, and those values that could be inferred from the submittal for CA, SI, and ERCW.

Assuming the individual system results are independent the calculated total ΔCDF value of 2.0E-7/reactor-year is well within the established acceptance guidelines (i.e., in Region III of Figure 4 of RG 1.174). The calculated total ΔLERF integrated risk analysis (IRA) value of 1.1E-8/reactor-year cited in the licensee's submittal and used to determine its acceptability does not include the total contribution from all valve groups. Rather, it is strictly the contribution from the IRA performed for the PA, WD, and RM systems. The ΔLERF values inferred for the ERCW and SI valve groups are substantially greater than the licensee's cited total ΔLERF(IRA) value. Further, the inferred SI ΔLERF value of 1.2E-7/reactor-year would place the licensee's results in Region II of Figure 4 of RG 1.174. For results in this region, the RG indicates some need for increased technical review and management attention and, thus, it is assumed that there would need to be a greater level of detail provided in the licensee's submittal.

Further, the licensee's calculations do not address the aggregate impact of the extension of all the selected valve test intervals that are proposed under the limited scope RI-IST Program. The aggregate impact of these test interval extensions may be greater than the individual system group impacts.

The staff finds that the licensee's evaluations of ΔCDF and ΔLERF are not complete, and therefore, it is not possible to develop a defensible conclusion as to acceptability.

4. CONCLUSIONS

The licensee provided some quantitative results for the impact on CDF and LERF at the system valve group level. However, the licensee's analyses do not address a number of plant conditions and configurations and do not address the aggregate impact of the test intervals being increased for all selected valves. In addition, the valve selection process, though based on recent performance history (i.e., last 3 years), does not address the safety significance of the selected valves and does not identify poor performing valves for which improved or enhanced test methods may be needed. Finally, the licensee's submittal does not adequately address traditional engineering considerations, such as defense-in-depth and safety margins.

The staff has identified a number of deficiencies associated with the licensee's submittal that could impact the overall conclusions and the staff finds that the licensee's risk-informed approach and results are incomplete and do not address some important elements of an acceptable risk-informed submittal. Therefore, the staff has determined that it is not possible to develop a defensible conclusion regarding the acceptability of the licensee's proposed limited scope RI-IST Program for the selected valves.

REFERENCES

1. Letter, dated April 27, 2000, from Pedro Salas (TVA, Licensing and Industry Affairs Manager) to NRC, regarding *Sequoyah Nuclear Plant (SQN) - Request for Approval of Relief from American Society of Mechanical Engineers (ASME) Code Requirements - Request for Relief RI-IST-1 - Risk Informed Inservice Testing - Valves*.

2. Letter, dated September 20, 2000, from Richard P. Correia (NRC, Office of Nuclear Reactor Regulation, Division of Licensing Project Management, Project Directorate II, Section 2, Chief) to Mr. J. A. Scalice (TVA, Chief Nuclear Officer and Executive Vice President), regarding *Relief from American Society of Mechanical Engineers (ASME) Code Requirements for Check Valve Inservice Testing Requirements at Sequoyah Nuclear Plant, Units 1 and 2 (TAC Nos. MA9882 and MA9883)*

3. NRC, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing*, RG 1.175, August 1998.

4. Letter, dated March 30, 2001, from Pedro Salas (TVA, Licensing and Industry Affairs Manager) to NRC, regarding *Sequoyah Nuclear Plant (SQN) - Withdrawl of Request for Relief from American Society of Mechanical Engineers (ASME) Code Requirements - Risk Informed Inservice Testing (RI-IST) Program (TAC Nos. MA9097 and MA9098)*.

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

Principal Contributor: Donald G. Harrison

Date: May 4, 2001

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. Jack A. Bailey
Vice President
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 10H
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. Robert J. Adney, General Manager
Nuclear Assurance
Tennessee Valley Authority
5M 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, Manager
Licensing 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 Harris
305 Pickel Road
Ten Mile, TN 37880