Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

September 30, 2013

10 CFR Part 54

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

Sequoyah Nuclear Plant, Units 1 and 2
Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

Subject: **Response to NRC Request for Additional Information Regarding the Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application, Sets 10 (B.1.23-2a), 11 (4.1-8a), and 12 (30-day) (TAC Nos. MF0481 and MF0482)**

References:

1. Letter to NRC, "Sequoyah Nuclear Plant, Units 1 and 2 License Renewal," dated January 7, 2013 (ADAMS Accession No. ML13024A004)
2. NRC Letter to TVA, "Requests for Additional Information for the Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application - Set 10," dated August 2, 2013 (ADAMS Accession No. ML 13204A257)
3. NRC Letter to TVA, "Requests for Additional Information for the Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application - Set 11," dated August 22, 2013 (ADAMS Accession No. ML 13224A126)
4. NRC Letter to TVA, "Requests for Additional Information for the Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application - Set 12," dated August 30, 2013 (ADAMS Accession No. ML 13238A244)

By letter dated January 7, 2013 (Reference 1), Tennessee Valley Authority (TVA) submitted an application to the Nuclear Regulatory Commission (NRC) to renew the operating licenses for the Sequoyah Nuclear Plant (SQN), Units 1 and 2. The request would extend the licenses for an additional 20 years beyond the current expiration date.

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By Reference 2, the NRC forwarded a request for additional information (RAI) labeled Set 10. The NRC License Renewal Project Manager, Mr. Richard Plasse, had given a verbal extension for RAI B.1.23-2a from that set until October 1, 2013. Enclosure 1 provides the response to RAI B.1.23-2a.

By Reference 3, the NRC forwarded an RAI labeled Set 11. The required date for responding to this RAI set is no later than October 21, 2013. However, Enclosure 1 provides the early response to RAI 4.1-8a.

By Reference 4, the NRC forwarded an RAI labeled Set 12. The required date for responding to this RAI set is no later than September 30, 2013. However, Mr. Plasse has given a verbal extension for RAI B.1.23-2b until October 29, 2013. Enclosure 2 provides the RAI responses for the rest of the Set 12 RAIs.

Enclosure 3 is an updated list of the regulatory commitments for license renewal.

Consistent with the standards set forth in 10 CFR 50.92(c), TVA has determined that the additional information, as provided in this letter, does not affect the no significant hazards considerations associated with the proposed application previously provided in Reference 1.

Please address any questions regarding this submittal to Henry Lee at (423) 843-4104.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 30th day of September 2013.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:
1. TVA Responses to NRC Request for Additional Information: Sets 10 (B.1.23-2a) and 11 (4.1-8a)
2. TVA Responses to NRC Request for Additional Information: Set 12 (30-day)
3. Regulatory Commitment List, Revision 8

cc (Enclosures):
NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

ENCLOSURE 1

Tennessee Valley Authority

Sequoyah Nuclear Plant, Units 1 and 2 License Renewal

TVA Responses to NRC Request for Additional Information: Sets 10 (B.1.23-2a) and 11 (4.1-8a)

Set 10: RAI B.1.23-2a

Background:

In its July 1, 2013, response to RAI B.1.23-2, the applicant addressed wear of the control rod drive mechanism (CRDM) nozzles resulting from interactions with the centering pads of the CRDM nozzle thermal sleeves. According to the applicant's analysis, the maximum wear depth will not exceed 0.05 inches based on design parameters and the assumption of uniform material properties and wear progression. On the basis of this analysis, the applicant stated that loss of material due to wear is not an aging effect requiring management for the CRDM nozzles.

Issue:

The applicant's analysis involves uncertainties due to unknown variations in local vibratory motions, residual stresses, and hardness levels of the CRDM nozzles, thermal sleeves, and centering pads. In addition, the LRA does not identify an inspection program to manage loss of material due to wear for the CRDM nozzles. Without inspections, the actual progression of the wear profiles cannot be well characterized and localized severe wear conditions cannot be excluded.

Request:

Justify why an inspection program is not necessary to confirm that wear is not impacting the reactor coolant pressure boundary function of the CRDM nozzles. Alternatively, identify an inspection program and justify why it will adequately manage loss of material due to wear for the CRDM nozzles.

TVA Response to RAI B.1.23-2a

CRDM inside diameter nozzle wear was evaluated by Westinghouse in a Sequoyah Nuclear Plant (SQN) plant specific CRDM adapter wear analysis (Reference 1). The analysis determined that a 0.050 inch wear groove depth in the control rod drive mechanism (CRDM) head adapters (also referred to as CRDM nozzles or housings), due to contact with the thermal sleeve centering pads, is a reasonable upper bound that is not expected to be exceeded through the PEO.

This maximum wear depth of 0.050 inches is based on the assumption that the 0.1075 inch thick thermal sleeve centering pads and the CRDM head adapter inside surface will wear at approximately the same rate, which is justified by the following.

1. Industry experience has shown wear on the outer surface of the 304 stainless steel thermal sleeve material where the thermal sleeve exits the CRDM head adapter. However, there were no reports of obvious or significant wear in the CRDM head adapter inside surface at or near the bottom of the head adapter.
2. The thermal sleeve centering pads are made of the same material as the thermal sleeve tube and are not hardened.
3. The thermal sleeve centering pads and CRDM head adapter inside surface have identical surface finishes.
4. Wear is related to local vibratory excitation and surface hardness. The minimum CRDM head adapter wall thickness criteria were developed based on high-cycle fatigue due to flow-induced vibration and pump-induced vibration loads. The flow-induced vibration loads are derived from the spray nozzle jet cross-flow velocities at the exposed portions of the thermal sleeves. Moments resulting from the pump-induced vibration were also considered. The specific hardness values of the sleeve, centering pads and CRDM head adapter are unknown, but similar grades of stainless steel and Inconel have similar hardness values (Rb ~90).
5. When contact between the thermal sleeve centering pads and head adapter inside surface occurs, the relatively small wear volume of the three centering pads is distributed over the relatively large area of the head adapter inside surface.
6. Stress intensity results calculated for CRDM head adapters were recalculated using reduced head adapter wall thickness (Reference 1). The head adapters were also evaluated for fatigue usage using the same reduced wall thickness and it was determined the usage would remain less than 1.

An Owners Group report (Reference 2) performed scoping evaluations of primary stress, stress intensity ranges and fatigue usage to determine the depth of wear in the CRDM head adapter that could be qualified per Section III of the ASME Code. Using enveloping loads and transient sets, the scoping evaluation demonstrated that the CRDM head adapter stress and fatigue usage were less than the ASME Code allowable with head adapter wear equal to a depth of 0.10-inch , twice the expected upper bound of wear. Wear depth of greater than 0.10 inch is not expected because the thermal sleeve centering pads that produce the wear in the CRDM head adapter will also wear to some degree. The very conservative assumptions used in this Owners Group scoping report provides further justification that even with bounding type assumptions ASME Code compliance could be demonstrated.

As a result, loss of material due to wear will not impact the reactor coolant pressure boundary function of the SQN CRDM head adapters, and wear inspections of the CRDM head adapters will not be required. Industry experience noted in the Owners Group report that is applicable to SQN Units 1 and 2, provides further justification for establishing the 0.050 inch wear groove depth as an upper bound. The Owners Group report described circumferential wear grooves in a CRDM head adapter at one four-loop reactor. This wear groove was measured to be 0.010 inches deep, and was located in the centermost head adapter where the lower centering pads are closest to the J-groove weld. As noted previously, the SQN plant specific analysis assumed a wear depth of 0.050 inches based on the assumption of equal wear rates which provides significant margin over the measured wear depth of 0.010 inches.

In addition to the above, as a part of the existing ASME Section XI Code Case N-729 augmented inspections for the CRDM head adapters, the two outermost concentric rows of penetration thermal sleeves are examined for evidence of material thinning, in accordance with Westinghouse Technical Bulletin TB-07. SQN continues to evaluate industry operating experience related to CRDM head adapter wear and initiatives to measure CRDM head adapter thickness.

References:

1. WCAP 16903P, Revision 0, April 2008, Addendum to Analytical Reports for Sequoyah Units 1 and 2 Reactor Vessels (CRDM Head Adapter Wear Justification)
2. WCAP 17725P, Revision 0-A, February 2013, Scoping Study to Determine the Feasibility of a Generic CRDM Housing Wear Evaluation

***Set 11:* RAI 4.1-8a**

Background:

By letter dated July 11, 2013, the applicant provided its responses to RAI 4.1-8, Parts 1 and 2, on whether the Updated Final Safety Analysis Report (UFSAR) Section 10.2.3 includes any plant turbine analyses that would need to be identified as TLAAs in accordance with requirements for identifying TLAAs in 10 CFR 54.21(c)(1). The staff has determined that the applicant's response to RAI 4.1-8, Part 1 provides adequate demonstration that the probabilistic analyses for the high pressure turbines (HPTs) and low pressure turbines (LPTs) do not need to be identified as TLAAs for the LRA.

RAI 4.1-8a Issue 1:

The applicant stated in its response to RAI 4.1-8, Part 2 that evaluation of stress corrosion cracking (SCC) in Westinghouse Report WSTG-1-NP (i.e., Reference 3 in the RAI response) is not a TLAA because it does not involve time-limited assumptions. However, SCC is identified in GALL Table IX.F as time-dependent aging mechanism, which implies that the analysis of SCC involves a time-limited assumption, unless demonstrated to the contrary. In contrast, the response to the RAI did not provide any reason why the analysis does not involve a time-limited assumption and therefore does not adequately demonstrate that the evaluation of SCC in the referenced Westinghouse analysis would not need to be identified as a TLAA for the LRA.

Request 1:

Explain how the analysis of SCC was performed in Westinghouse Technical Report No. WSTG-1-NP (i.e., Ref. 3 in the response to RAI 4.1-8). Based on this explanation, clarify why the analysis of SCC in the report is not considered to involve time-limited assumptions. Based on your response, provide your basis (i.e., justify) why the analysis of SCC in the referenced Westinghouse report does not need to be identified as a TLAA, when compared to the six criteria for defining an analysis as a TLAA in 10 CFR 54.3(a).

TVA Response to RAI 4.1-8a, Request 1

Westinghouse Technical Report No. WSTG-1-NP predicts probability of failure based on 1) time since the last inspection and 2) stress corrosion crack growth rate. The results are shown on Figure 9 of the WSTG-1-NP "Probability of Missile Generation as a Function of Inspection Interval Year." The probability of a given nuclear turbine experiencing a low-pressure disc rupture due to stress corrosion cracking on the bore or in a keyway of a disc was calculated. A 40-year time frame was not used as an input to this Westinghouse analysis; rather, only the time since the last inspection was used. Therefore, the Westinghouse turbine failure analyses based on stress corrosion cracking (SCC) do not meet the TLAA definition because they do not involve time-limited assumptions defined by the current term of operation, for example, 40 years.

RAI 4.1-8a Issue 2:

The applicant stated in its response to RAI 4.1-8, Part 2 that "no fatigue-based analysis was required or used in the turbine missile evaluation." However, UFSAR Section 10.2.3 (i.e., UFSAR page 10.2-9) makes the following statement:

> *Prior to 1980, the Westinghouse missile probabilities and energies analyses were directed primarily at missile generation due to destructive overspeed. Fatigue of the rotating elements due to speed cycling was also considered as a missile generation mechanism in these earlier analyses. These earlier Westinghouse analyses indicated that the probabilities of missile generation due to fatigue and destructive overspeed were very low in comparison to the probability estimated by Bush. The Bush probability (1 x 100-4 missile producing disintegrations per turbine operating year) was chosen for the original Sequoyah missile hazard evaluation in order to provide a very liberal margin of safety.*

Based on this UFSAR statement, it appears that the Westinghouse fatigue analyses of the LPT rotating elements were used to confirm the missile generation probabilities of the Bush studies (as referenced in the UFSAR and response to RAI 4.1-8, Part 1) that were used for the LPTs. It is not evident why these Westinghouse analyses would not need to be identified as TLAAs for the LRA.

Request 2:

1. *Identify the Westinghouse fatigue analyses that were referenced on UFSAR page 10.2-9 and performed in analysis of the LPT rotating elements.*
2. *Explain how the assessment of fatigue was performed in these analyses.*
3. *Provide your basis (i.e., justify) why the stated Westinghouse fatigue analyses of the LPT rotating elements would not need to be identified as TLAAs for the LRA, when compared to the six criteria for defining an analysis as a TLAA in 10 CFR 54.3(a)*

TVA Response to RAI 4.1-8a, Request 2

The paragraph from UFSAR Section 10.2.3 is a general statement about the results of "earlier analyses" that were used prior to 1980. The paragraph states that the Bush value was very conservative in comparison. This comparison is explained further in the UFSAR paragraph following the UFSAR Section 10.2.3 paragraph. (Note the cited UFSAR paragraph has a typographical error- 100-4 should be 10^{-4} consistent with values on UFSAR pages 10.2-14 and 10.2-15 and in the Bush report. TVA will correct the error in the Corrective Action Program.)

The paragraph following the cited paragraph of UFSAR Section 10.2.3 (i.e., UFSAR page 10.2-9) identifies the probability due to fatigue and destructive overspeed. The conclusion was that the probabilities are very low when compared to the probability recommended by Bush and compared to the probability of missile generation due to SCC. This paragraph states for fatigue:

> These new fatigue missile generation probabilities are six to seven orders of magnitude lower than the maximum allowable turbine missile generation probability and thus are insignificant.

The same paragraph states for destructive overspeed:

> The probability of missile generation due to SCC at design overspeed conditions (120 percent of rated speed) is two orders of magnitude lower than the probability of missile generation due to SCC at rated speed. Consequently, the probability of missile generation at Sequoyah (due to all failure mechanisms) is, for analysis purposes, approximately equal to the probability of missile generation due to SCC at rated speed.

This paragraph notes that the fatigue missile generation probability was insignificant. These probability calculations were not used for the calculated probability for Bush or for the probability calculation for missile generation from SCC. Thus, they do not meet element 5 of the TLAA definition because they did not provide the basis for conclusions related to the capability of the component to perform its intended functions.

ENCLOSURE 2

Tennessee Valley Authority

Sequoyah Nuclear Plant, Units 1 and 2 License Renewal

TVA Responses to NRC Request for Additional Information: Set 12 (30-day)

RAI 4.3.1-2

Background:

LRA Table 4.3-1 and 4.3-2 lists the projected and analyzed transient cycles for Unit 1 and Unit 2 respectively.

RAI 4.3.1-2 Issue 1:

In LRA Tables 4.3-1 and 4.3-2, the applicant does not identify any past operating experience (i.e., through operations as of November 1, 2011 for the units) for the primary side leak test transient. Specifically, the staff seeks justification on why the LRA does not list at least the following cycle number in the "Cycles as of Nov. 1, 2011" column of the tables for the primary side leak test, a number of past primary side system leak test occurrences equivalent to the total numbers of system leak tests that were performed over the past 31 years for Unit 1 and 30 years for Unit 2 in accordance with the ASME Code Section XI, Examination Category B-P primary side system leak test requirements.

Request 1:

Specifically, for the primary side leak test transient, provide your basis why the "Cycles as of Nov. 1, 2011" column in the tables do not cite a value that is at least as conservative as the total number of primary side leak test performed over the past 31 years for Unit 1 and 30 years for Unit 2 in accordance with the ASME Code Section XI, Examination Category B-P system leak test requirements and possibly during past maintenance outages.

TVA Response to Request 1

The primary side leak test cycles are specific to the analyses for the steam generators (SGs). As stated in LRA B.1.39, the SQN Unit 1 and Unit 2 SGs were replaced in 2003 and 2012, respectively. Thus, the primary side leak test cycles were reset to zero upon replacement of the SGs for both units. The analyses qualified the replacement SGs for 50 cycles for the primary side leak test. The test has not been performed since the installation of the replacement SGs, so the current cycle count is zero.

The primary side leak test transient is defined as raising the primary pressure to 2485 psig and maintaining the differential pressure across the SG tube sheet to less than 1600 psid. However, the allowable test pressure per ASME Section XI is the normal operating pressure of 2235 psig; this test pressure is used for SQN Units 1 and 2. Because the leak test is performed at normal operating pressure, the primary side leak test transient that pressurizes to 2485 psig is not required. The Fatigue Monitoring Program will track the cycles if they are performed and ensure the cycles remain below the allowable number.

RAI 4.3.1-2 Issue 2:

Since the applicant used the 60-year transient projections to support the disposition of the time-limited aging analyses (TLAAs) evaluated in LRA Sections 4.7.3, the staff requires additional information to determine whether the methodology used in the cycle projection methodology is appropriate.

Request 2:

Justify why LRA Tables 4.3-1 and 4.3-2 do not provide any 60-year cycle projection values for the following design basis transients: (a) the "½ safe shutdown earthquake" transient; (b) the low-temperature overpressure protection actuation; (c) the secondary side hydrostatic test condition transient; and (d) the primary side leak test transient.

TVA Response to Request 2

The projection method used is based on the cycles that have occurred to determine a rate and then uses that rate to determine a projected value for 60 years. Because there have been zero "½ safe shutdown earthquakes," "low-temperature overpressure protection actuations" and "secondary side hydrostatic test condition" transients, the rate experienced per year is zero.

A rate of zero cycles per year multiplied by 60 years results in a projection of zero. A cycle projection of zero for the primary side leak test transient is explained in the TVA response to Request 1 of this RAI.

The projected values in the LRA Tables 4.3-1 and 4.3-2 are information-only values used for comparison purposes. The projected values do not change the allowable numbers of cycles for the components and are not new cycle limit values. The allowable numbers of cycles remain the same as the values used in the analyses, and are greater than the cycles that are expected through the period of extended operation (PEO).

The Fatigue Monitoring Program will track the actual cycles and ensure the number of cycles remain below the allowable number.

RAI 4.3.1-3

Background:

LRA Section 4.3.1.4 provides the applicant's metal fatigue TLAAs for the replacement steam generator (SG) components. The applicant provides its cumulative usage factor (CUF) values for these SG components in LRA Table 4.3-6, including the CUF value for the SG U-bend support tree at Unit 1.

Issue:

The LRA indicates that a fatigue analysis was performed for the SG U-bend support tree at Unit 1, but not for the same component at Unit 2.

Request:

Provide the basis why the SG U-bend support tree for Unit 2 had not been subjected to a metal fatigue analysis in the manner that the SG U-bend support tree for Unit 1 had been analyzed for fatigue.

TVA Response to RAI 4.3.1-3

The design of the SQN Unit 1 and Unit 2 replacement SGs are similar, but not identical.

The Unit 2 replacement SG design includes an improvement in the upper bundle tube support structure. This improvement results in calculated stresses below the fatigue endurance limit; therefore, a cumulative usage factor (CUF) value was not calculated for this location on Unit 2.

RAI 4.3.1-4

Background:

In LRA Section 4.3.1.6, the applicant identifies that the reactor coolant pump (RCP) design includes RCP thermowells that received a CUF analysis, and that the CUF values for the RCP thermowells are negligible. In LRA Section 4.3.1.7, the applicant identifies that the reactor coolant system (RCS) hot legs and cold legs were modified to include thermowells and that the fatigue waiver analyses for the thermowells in the RCS hot legs and cold legs were TLAAs for the LRA.

Issue:

The staff cannot determine whether the RCP thermowells referred to in LRA Section 4.3.1.6 are the same component as any of the thermowells that were referred to in LRA Section 4.3.1.7 for the hot leg and cold leg designs.

Request:

Clarify whether the RCP thermowells referred to in LRA Section 4.3.1.6 are the same as any of the thermowells that were referenced in LRA Section 4.3.1.7 for the RCS hot legs and cold legs. Justify why the current licensing basis (CLB) for the thermowells in the RCS hot legs and cold legs would not need to have included fatigue analyses when a fatigue analysis was required as part of the CLB for the RCP thermowells. Revise LRA Appendix A as appropriate based on the response.

TVA Response to RAI 4.3.1-4

The thermowells on the reactor coolant pumps (RCPs) referred to in LRA Section 4.3.1.6 are part of the shaft seal assembly of the RCPs and are different components than the thermowells in the RCS hot legs and cold legs referred to in LRA Section 4.3.1.7.

The ASME Section III analysis of the RCP thermowells determined more than 10^6 cycles were allowed. This result is summarized in the analysis as a CUF of "negligible."

When the resistance temperature detector bypass piping was removed and direct sensing resistance temperature detectors installed on the hot and cold legs, thermowells were installed. UFSAR Sections 5.5.3.2 and 5.6 provide additional details of the configuration. An analysis determined that the thermowells were exempt from a detailed fatigue analysis (i.e., no CUF was calculated) because the provisions of the applicable design code section (1983 ASME NB-3222.4(d)) were satisfied. This exemption is based on the reactor coolant system (RCS) transients shown in LRA Tables 4.3-1 and 4.3-2 and is, therefore, considered a TLAA as identified in LRA Section 4.3.1.7.

Both of these analyses verified the acceptability of the associated thermowells for fatigue.

No change to LRA Appendix A is necessary.

RAI 4.3.1-5

Background:

LRA Section 4.3.1.7 includes the implicit fatigue TLAAs for the Safety Class 1 or Class A piping systems that were designed to the standards in the USAS B31.1 design code.

Issue:

The staff noted that the applicant did not identify which of the design basis transients in LRA Table 4.3-1 or 4.3-2 constituted actual full thermal range transients for the implicit fatigue analysis that was performed for the Safety Class 1/Class A piping systems that were designed to the USAS B31.1 design code requirements, or the type of piping, piping components, piping elements that were included within the scope of the analyses for these systems.

Request:

Identify all Safety Class 1 or Class A systems (including Class 1 or Class A portions of interfacing systems to the RCS), and the piping, piping components, and piping elements in these systems, that were within the scope of the applicable implicit fatigue analysis requirements in the USAS B31.1 design code. For these systems, identify the design basis transients that constitute "full thermal range" transients for the implicit fatigue analyses of the systems. Justify that the total number of occurrences of those "full thermal range" transients remain less than 7000. Revise LRA Appendix A as appropriate based on the response.

TVA Response to RAI 4.3.1-5

As shown in UFSAR Table 3.2.2-2 and discussed in UFSAR Section 5.5.3, the original design analyses for the RCS piping was in accordance with United States of America Standard (USAS) B31.1. The piping, piping components, and piping elements analyzed in accordance with USAS B31.1 design code for SQN Units 1 and 2 include the piping components in the RCS loops and the Class 1 components that connect to the RCS pressure boundary including portions of the safety injection system (SIS), residual heat removal (RHR) system, and chemical and volume control system (CVCS). USAS B31.1 states that "Piping as used in this Code includes pipe, flanges, bolting, gaskets, valves, relief devices, fittings and the pressure retaining parts of other components." The USAS B31.1 definitions further indicate that the term "pipe" includes "tubing." For further information, see UFSAR Section 5.2.1.

The Class 1 piping, piping components and piping elements are identified in LRA Table 3.1.2-3 "Reactor Coolant Pressure Boundary" and includes portions of the SIS, RHR system, and CVCS. The Class 1 or Class A boundary is shown on the following LRA drawings.

System (System Code)	LRA Drawing(s)
Reactor Coolant System (68)	LRA-1,2-47W813-1
Safety Injection System (63)	LRA-1-47W811-1 and LRA-2-47W811-1
Residual Heat Removal System (74)	LRA-1, 2-47W810-1
Chemical & Volume Control System (62)	LRA-1-47W809-1 and LRA-2-47W809-1

The RCS piping and system piping adjacent to the main coolant loops would be heated up when the RCS is heated up. As shown in LRA Tables 4.3-1 and 4.3-2, plant heatups are limited to less than 200 cycles and specific system details are provided below:

- Portions of the SISs that are normally at elevated temperatures during normal plant operation would be cooled if a safety injection occurred (limited to 110 cycles).
- The piping in the RHR loop that is not close enough to the RCS main loop piping to be at elevated temperatures during normal plant operation could be heated above the fatigue threshold when the RHR system is placed in service during a plant cooldown (limited to less than 200 cycles).
- Portions of the CVCS system can experience thermal cycles if the CVCS flow is terminated long enough for the piping to cool. CVCS thermal cycles for plant heatup and cooldown are limited to 200 cycles. CVCS flow termination may occur during plant transients such as loss of load without trip (80 cycles), loss of AC Power (40 cycles), loss of flow in one RCS loop (80 cycles), and reactor trips (400 cycles). SQN CVCS transients are tracked and would result in no more than 600 total cycles (80+40+80+400). See LRA Tables 4.3-1 and 4.3-2.
- The pressurizer spray line can experience a significant temperature transient if auxiliary spray is initiated (limited to a total of 10 cycles).

As shown above, the total number of cycles experienced by the RCS components will remain well below the 7000 cycles of the implicit fatigue analysis of ANSI B31.1 through the PEO.

No change is necessary to LRA Appendix A.

RAI 4.3.1-6

Background:

LRA Section 4.3.1.7 includes the metal fatigue TLAA for the pressurizer surge lines. The applicant states that it will use the cycle monitoring activities and the periodic CUF update activities of the Fatigue Monitoring Program to accept the TLAA for the pressurizer surge lines in accordance with the criterion in 10 CFR 54.21(c)(1)(iii) and to manage the impacts of cracking by fatigue on the intended pressure boundary function of the surge lines during the period of extended operation.

The staff noted that the NRC addressed the impact of thermal stratification stresses on the pressure boundary functions of pressurizer surge lines in NRC Bulletin (BL) 88-11, "Pressurizer Surge Line Thermal Stratification" (December 20, 1988). The staff noted that the applicant addressed the issues and requests that were identified in BL 88-11 in the following four TVA letters to the NRC:

1. *TVA Letter of April 18, 1989 (NRC Accession No. 8905010150 and Microfiche 49554, Fiche Pages 334-338)*
2. *TVA Letter of May 26, 1989 (NRC Accession No. 8906020225 and Microfiche 49988, Fiche Pages 300-306)*
3. *TVA Letter of June 22, 1989 (NRC Accession No. 8907050132 and Microfiche 50401 Fiche Pages 103-132)*
4. *TVA Letter of Sept. 6, 1989 (NRC Accession No. 89009120190 and Microfiche 51179, Fiche Pages 71-72)*

Issue:

The program elements of the applicant's Fatigue Monitoring Program includes steps to update the respective CUF analysis on an as needed basis, as based on the results of the program's cycle counting activities for the transients that were assumed for in the analysis for the pressurizer surge lines. It is not evident to the staff on whether such potential updates of the CUF analysis for the pressurizer surge lines will continue to address potential impact of thermal stratification stresses on the CUF results for the updated analysis.

Request:

Clarify whether potential updates of the CUF analysis for the pressurizer surge line under the Fatigue Monitoring Program would continue to address potential impacts of thermal stratification stresses on the results of the CUF analysis. If yes, clarify how the Fatigue Monitoring Program will be used to address potential impacts of thermal stratification stresses on the results of the updated CUF analysis. If not, justify why any updates of the CUF analysis for the pressurizer surge lines would not need to address potential impacts of thermal stratification stresses on the fatigue analysis results for the pressurizer surge lines. Revise LRA Appendix A as appropriate based on the response.

TVA Response to RAI 4.3.1-6

Under the Fatigue Monitoring Program described in LRA B.1.11, potential updates of the CUF analysis for the pressurizer surge line would address potential impacts of thermal stratification stresses on the results of the CUF analysis as described in the following paragraphs.

In response to NRC Bulletin (BL) 88-11, a site-specific calculation was generated for a fatigue life assessment of the RCS pressurizer considering insurge/outsurge transients which may occur during plant heatup and cooldown. Insurge/outsurge events were determined by examining real plant data during pressurizer heatups and cooldowns. A conservative spectrum of insurge/outsurge events defining the severity (temperature differential) and the number of occurrences per heatup or cooldown cycle was developed. The overall number of insurge/outsurge events was then determined by prorating the insurge/outsurge spectrum to the number of heatups and cooldowns during plant life. For periods other than heatup and cooldown, the system differential temperature is generally less than 150°F, and when considering real plant data behavior, the effect of any insurge/outsurge cycles on fatigue of the pressurizer surge nozzle was judged to be negligible or below the endurance limit for each cycle. The fatigue effect of insurge/outsurge cycling is adequately managed by counting the number of heatup and cooldown events..

The fatigue usage calculated for insurge/outsurge transients assumes a total of 200 heatups and cooldowns. See LRA Tables 4.3-1 and 4.3-2. The resulting fatigue usage for 200 heatup and cooldown cycles is added to the cumulative fatigue usage computed by Westinghouse in the original fatigue analysis for the pressurizer based on the transients identified in Table 4.3-1 and Table 4.3-2 for SQN Unit 1 and Unit 2, respectively.

LRA Table 4.3-5 lists the calculated fatigue usage for the pressurizer surge nozzle. The calculated fatigue usage is the sum of the usage calculated for all the original transients identified in Tables 4.3-1 or 4.3-2 plus the additional usage due to insurge/outsurge transients from a total of 200 heatups and cooldowns. If the pressurizer surge nozzle cycle limits identified in Tables 4.3-1 and 4.3-2 are approached, then additional fatigue usage due to insurges and outsurges will again be added to calculate the total fatigue usage. As shown in Tables 4.3-1 and 4.3-2, the projected heatup and cooldown cycles through the PEO are less than the design value of 200. The Fatigue Monitoring Program will continue to track the number of plant heatups and cooldowns.

No change to LRA Appendix A is necessary.

RAI 4.3.1-7

Background:

LRA Section 4.3.1.7 identifies that thermowells were installed and that the cycle-based fatigue waiver analyses for the thermowells, as performed in accordance with ASME Section III fatigue waiver provisions, are TLAAs for the LRA. In this section of the LRA, the applicant states that the cycle counting activities of LRA AMP B.1.11, "Fatigue Monitoring Program," will be used to accept this TLAA in accordance with the requirement in 10 CFR 54.21(c)(1)(iii) and to manage the impacts of fatigue on the intended reactor coolant pressure boundary function of the thermowells.

Issue:

The scope of the current program description and program elements in GALL AMP X.M1, "Fatigue Monitoring Program," only includes cycle counting and monitoring bases for those analyses that are defined as cycle-based cumulative usage factor (CUF) analyses. The program has not been extended by the applicant to include program element criteria for using the cycle counting bases to monitor against other types of cycle-based analyses, such as cycle-based ASME fatigue waiver analyses or cycle-based flaw tolerance or fracture mechanics analyses.

To extend the scope of AMP B.1.11, Fatigue Monitoring Program, to the monitoring of the RCS transients that have been analyzed in applicable ASME Section III fatigue waiver analyses, the applicant may need to enhance the program elements including, but not limited to, "scope of program," "detection of aging effects," "monitoring and trending," and "acceptance criteria" program appropriately to account for the fact that the program is also being credited for monitoring of the design transients that have been assumed in applicable ASME Section III fatigue waiver analyses.

Request:

Provide your basis for using the Fatigue Monitoring Program to accept the fatigue waiver analysis for the RCS hot-leg and cold-leg thermowells in accordance with 10 CFR 54.21(c)(1)(iii), without including any enhancements of program elements to account for cycle count monitoring activities against these types of analyses. Revise LRA Appendix A as appropriate based on the response.

TVA Response to RAI 4.3.1-7

The Fatigue Monitoring Program described in LRA Section B.1.11 governs cycle counting of RCS heatups and cooldowns. The thermowells installed to replace the resistance temperature detector system were qualified to ASME Section III. The thermowells were exempt from a detailed fatigue analysis (i.e., no CUF was calculated) because the 1983 ASME NB-3222.4(d) requirements were satisfied. The exemption was based on the number of cycles the thermowells would experience during 200 plant heatups and cool-downs.

The Fatigue Monitoring Program manages the fatigue of the thermowells in accordance with 10 CFR 54.21(c)(1)(iii) because it tracks plant heatups and cool-downs.

As described in LRA Sections A.1.11 and B.1.11, Fatigue Monitoring Program, the program is credited for addressing applicable fatigue exemptions or waivers. The Fatigue Monitoring

Program procedures are updated in the event the number of thermowell heatup and cooldowns approaches the cycle limit assumed in the fatigue analysis in accordance with 1983 ASME NB-3222.4(d) requirements. The changes to LRA Section A.1.11 and B.1.11 follow, with additions underlined.

LRA Section A.1.11

- Revise Fatigue Monitoring Program procedures to provide updates of the fatigue usage calculations and cycle-based fatigue waiver evaluations on an as-needed basis if an allowable cycle limit is approached, or in a case where a transient definition has been changed, unanticipated new thermal events are discovered, or the geometry of components has been modified.

LRA Section B.1.11

Element Affected	Enhancement
4. Detection of Aging Effect	Revise Fatigue Monitoring Program procedures to provide updates of the fatigue usage calculations and cycle-based fatigue waiver evaluations on an as-needed basis if an allowable cycle limit is approached, or in a case where a transient definition has been changed, unanticipated new thermal events are discovered, or the geometry of components has been modified.

Commitment 7.D has been revised with additions underlined.

RAI 4.3.1-8

Background:

In LRA Table 4.3-12, the applicant provides the CUF-Fen results for pressurizer surge lines, including the low-alloy steel pressurizer surge nozzles with the CUF values of 0.49471 and 0.36634, for Units 1 and Unit 2 respectively. Both the USAR and LRA Table 3.1.2-3 identify that the pressurizer surge nozzle-to-safe end welds are made from Alloy 82/182 Inconel materials.

Issue:

It is not clear to the staff whether the pressurizer surge nozzle-to-safe end welds were considered as part of the fatigue analysis for the pressurizer surge nozzles or a separate CUF value was calculated for the pressurizer surge nozzle-to-safe end welds.

Request:

Clarify whether the pressurizer surge nozzle-to-safe end welds were considered to be within the scope of the fatigue analysis for the pressurizer surge nozzles.

If the answer to this request is yes, justify why the environmentally-assisted fatigue calculation that was performed on the pressurizer surge nozzle using the methodology in NUREG/CR-6583 for low-alloy steel components would be an acceptable basis for assessing environmentally-assisted fatigue in the pressurizer surge nozzle-to-safe end welds, which are made from nickel alloy materials.

If the answer to this request is no, clarify whether the pressurizer surge nozzle-to-safe end welds are in contact with the reactor coolant environment and how the effects of reactor coolant environment on the component fatigue life of the pressurizer surge nozzle-to-safe end welds will be managed during the period of extended operation.

TVA Response to RAI 4.3.1-8

The pressurizer surge nozzle-to-safe end weld was originally included in the fatigue analysis.

This weld is in contact with the reactor coolant environment; however, a full structural weld overlay is now installed over this weld assuming a through-wall defect has penetrated 360 degrees of the pipe circumference. Therefore, the pressurizer surge nozzle-to-safe end weld is now subject to flaw growth evaluation under ASME Section XI as opposed to fatigue analysis per Section III.

As identified in LRA Section 4.3.1.3, a flaw growth analysis, used to determine an appropriate inspection interval, has been prepared for the nickel-alloy weld in place of the original fatigue evaluation that had calculated a CUF.

RAI 4.3.2-2

Background:

LRA Section 4.3.2 identifies that an ASME Section III fatigue waiver was performed on the residual heat removal (RHR) heat exchangers and that the fatigue waiver analysis is a TLAA for the LRA. In this section of the LRA, the applicant states that the cycle counting activities of LRA AMP B.1.11, "Fatigue Monitoring Program," will be used to accept this TLAA in accordance with the requirement in 10 CFR 54.21(c)(1)(iii) and to manage the impacts of fatigue on the intended reactor coolant pressure boundary function of the RHR exchangers and to ensure that the fatigue waiver analysis for the RHR heat exchanges will remain valid for the period of extended operation.

Issue:

The scope of the current program description and program elements in GALL AMP X.M1, "Fatigue Monitoring Program," only includes cycle-counting and monitoring bases for those analyses that are defined as cumulative usage factor (CUF) analyses. The program has not been extended by the applicant to include program element criteria for using the cycle counting-bases to monitor against other types of cycle-based analyses, such as cycle-based ASME fatigue waiver analyses.

To extend the scope of AMP B.1.11, Fatigue Monitoring Program, to the monitoring of the RCS transients that have been analyzed for in applicable ASME Section III fatigue waiver analyses, the applicant may need to enhance the program elements including, but not limited to, "scope of program," "detection of aging effects," "monitoring and trending," and "acceptance criteria" program appropriately to account for the fact that the program is also being credited for monitoring of the design transients that have been assumed in applicable ASME Section III fatigue waiver analyses.

Request:

Provide the basis for using the Fatigue Monitoring Program to accept the fatigue waiver analysis for the RHR heat exchangers in accordance with 10 CFR 54.21(c)(1)(iii), without including any enhancements of the program elements to account for cycle-count monitoring activities against these types of analyses. Revise LRA Appendix A as appropriate based on the response.

TVA Response to RAI 4.3.2-2

The Fatigue Monitoring Program described in LRA Section B.1.11 performs cycle counting of the RCS heatups and cooldowns. The RHR heat exchangers were evaluated for fatigue and determined to meet the conditions for a cycle-based fatigue waiver in accordance with ASME Section III Paragraph N-415-1. The exemption is based on cycles the heat exchangers would experience during 200 plant heatups and cooldowns.

The Fatigue Monitoring Program manages the fatigue of the RHR heat exchangers in accordance with 10 CFR 54.21(c)(1)(iii) because it tracks plant heatups and cooldowns.

As described in LRA Sections A.1.11 and B.1.11, the Fatigue Monitoring Program is credited for addressing applicable fatigue exemptions or waivers. The Fatigue Monitoring Program provides for updates of the fatigue waiver evaluation in the event the number of RHR heat exchanger heatups or cooldowns approaches the cycle limit assumed in the fatigue waiver evaluation in accordance with Paragraph N-415-1 of ASME Section III.

The changes to LRA Section A.1.11 and B.1.11 are provided in the response to RAI 4.3.1-7 to indicate that cycle-based fatigue waiver evaluations will be updated as necessary if an allowable cycle limit is approached.

RAI 4.3.2-3

Background:

LRA Section 4.3.2.3 indicates that the CLB includes metal fatigue analyses for the heat exchangers in the chemical and volume control systems (CVCS) and fatigue waiver analyses for the RHR heat exchangers.

Issue:

During the staff's safety audit (March 18-22, 2013) of the aging management program (AMP) for mechanical systems, the staff noted the CLB includes metal fatigue analyses for the letdown heat exchangers and excessive letdown heat exchangers. However, the applicant has not justified why these fatigue analyses would not need to be identified as TLAAs, when compared to the six criteria in 10 CFR 54.3 for defining a plant analysis as a TLAA.

Request:

1. *Clarify how the fatigue analyses for the letdown heat exchangers and excessive letdown heat exchangers compare to the six criteria for TLAAs in 10 CFR 54.3.*

2. *Based on the response to Part a., clarify and justify whether the fatigue analyses for the letdown heat exchangers and excessive letdown heat exchangers need to be identified as a TLAAs in accordance with requirement in 10 CFR 54.21(c)(1). If the analyses need to be identified as a TLAAs, amend the LRA accordingly and provide the basis for dispositioning the TLAAs in accordance with 10 CFR 54.21(c)(1)(i), (ii), or (iii). Revise LRA Appendix A as appropriate based on the response.*

3. *Identify whether the CLB includes any other metal fatigue analyses or fatigue waiver analyses for Non-Safety Class 1/Non-Safety Class A heat exchanger components at the plant.*

4. *If it is determined that the CLB does include additional metal fatigue analyses or fatigue waiver analyses for heat exchanger components, identify each component-specific analysis that was performed as part of the CLB and justify why the applicable analysis would not need to be identified as TLAA in accordance with 10 CFR 54.21(c)(1).*

TVA Response to RAI 4.3.2-3

Response to Requests 1 and 2

No fatigue analyses for the letdown heat exchangers and excess letdown heat exchangers were identified. As shown in UFSAR Table 3.2.1-2, the letdown heat exchangers and excess letdown heat exchangers are Safety Class B on the tube side and Safety Class C on the shell side. The UFSAR table identifies the applicable ASME Code as Section III Class C for the tube side and Section VIII for the shell side. The ASME Code Sections III and VIII do not require fatigue analyses for these heat exchangers. No other analyses were identified that meet the definition of TLAA for the letdown heat exchangers and excess letdown heat exchangers.

Response to Requests 3 and 4

LRA Section 4.3.2.3 identifies the metal fatigue analyses for the CVCS regenerative heat exchangers and the fatigue waiver analyses for the RHR heat exchangers. There were no other analyses identified for the non-Safety Class 1/non-Safety Class A heat exchanger components. Therefore, no LRA change is necessary.

RAI 3.5.1-88

Background:

LRA Table 3.5.1, item 3.5.1-88, states that vibration, flexing of the joint, cyclic shear loads, thermal cycles and other causes can cause partial self-loosening of a fastener; however, these causes of loosening are minor contributors in structural steel and steel component threaded connections and are eliminated by initial preload bolt torquing. The LRA further states that SQN uses site procedures and manufacturer recommendations to provide guidance for proper torquing of nuts and bolts used in structural applications. Therefore, loss of preload due to self-loosening is not an aging effect requiring management for structural steel and steel component threaded fasteners within the scope of license renewal.

Issue:

The Structures Monitoring Program described in the GALL Report, which is an acceptable program to manage the loss of preload due to self-loosening for these components, not only considers the initial preload bolt torquing in the "preventive actions" program element, but also recommends inspection of structural bolting for loose bolts, missing or loose nuts, and other conditions indicative of loss of preload in the "parameters monitored or inspected" program element. The staff notes that the Structures Monitoring Program described in LRA Section B.1.40 has been enhanced to include the inspection of structural bolting for loose or missing nuts and to revise procedures to follow parameters to be monitored or inspected based on ANSI/ASCE 11, "Guideline for Structural Condition Assessment of Existing Buildings, American Society of Civil Engineers."

ANSI/ASCE 11, Section 3.3.2.6, "Physical Conditions of Connectors," and "3.3.3 Test Methods," provides guidelines for the inspection of the condition and tightness of the bolts which in addition to visual examination/observation include "physical assistance such as cleaning, scraping, and sounding" to establish the existence of snug fit "under some positive compressive force."

Based on the above, the staff's position is that the potential loss of preload due to self-loosening from vibration, flexing of the joint, cyclic shear loads, thermal cycles and other causes is an aging effect requiring management.

Request:

Provide the staff with sufficient technical basis for concluding loss of preload due to self-loosening is not an aging effect requiring management, or identify an aging management program to manage this aging effect.

TVA Response to RAI 3.5.1-88

Loss of preload due to self-loosening of structural bolting will be addressed as an aging effect requiring management for structural bolting.

The changes to LRA Table 3.5.1 Item 3.5.1-88 and Table 3.5.2-4 follow with additions underlined and deletions lined through.

Table 3.5.1: Structures and Component Supports					
Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Safety-Related and Other Structures; and Component Supports					
3.5.1-88	Structural bolting	Loss of preload due to self-loosening	Structures Monitoring Program	No	~~Vibration, flexing of the joint, cyclic shear loads, thermal cycles and other causes can cause partial self-loosening of a fastener. These causes of loosening are minor contributors in structural steel and steel component threaded connections and are eliminated by initial preload bolt torquing. SQN uses site procedures and manufacturer recommendations to provide guidance for proper torquing of nuts and bolts used in structural applications. Additionally, SQN site operating experience has not shown self-loosening of structural bolting used in SQN. Therefore, loss of preload due to self-loosing is not an aging effect requiring management for structural steel and steel component threaded fasteners within the scope of license renewal.~~ Consistent with NUREG-1801. The Structures Monitoring Program manages the listed aging effect.

Table 3.5.2-4
Bulk Commodities
Summary of Aging Management Evaluation

Table 3.5.2-4: Bulk Commodities								
Structure and/or Component or Commodity	**Intended Function**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801 Item**	**Table 1 Item**	**Notes**
Structural bolting: Structural steel and miscellaneous steel connections, including high strength bolting (decking, grating, handrails, ladders, platforms, stairs, vents and louvers, framing steel, etc.)	SNS, SRE, SSR	Carbon steel Galvanized steel	Air – indoor uncontrolled or Air – outdoor or Air with borated water leakage	Loss of preload	Structures Monitoring	III.A1.TP-261 III.A3.TP-261 III.A4.TP-261 III.A5.TP-261 III.A6.TP-261	3.5.1-88	A
Structural bolting	SNS, SRE, SSR	Carbon steel Galvanized steel Stainless steel	Air – indoor uncontrolled or Air – outdoor or Air with borated water leakage	Loss of preload	Structures Monitoring	III.A1.TP-261 III.A3.TP-261 III.A4.TP-261 III.A5.TP-261 III.A6.TP-261	3.5.1-88	A

RAI 3.5.1-2

Background:

SRP-LR Table 3.5-1 (sic, 3.5.1) includes line items for aging effects for accessible concrete areas that do not require further evaluation but recommend GALL Report AMPs to manage the effects of aging. In the Discussion column for several LRA Table 3.5-1(sic, 3.5.1)items, the applicant stated that the listed aging effects for the SQN steel containment vessel (SCV) concrete basemat do not require management at SQN. The discussion further states that SQN concrete is designed and constructed in a way that would prevent the effect of this aging from occurring and that aging effects are not significant for accessible areas.

For inaccessible areas associated with the listed aging effects, the applicant's response to RAI 3.5.1-1 stated that SQN is enhancing the Structures Monitoring Program (SMP) to require inspections of inaccessible areas in environments where observed conditions in accessible areas exposed to the same environment indicate that significant degradation is occurring.

Issue:

The staff does not agree that the aging effects associated with accessible areas of concrete do not require management. Regardless of the design and construction of the concrete, the staff believes all aging effects could occur in accessible and inaccessible areas and, therefore, require management. The discussion in the LRA states that the components are included in the SMP to confirm the absence of these aging effects; however, the associated line items do not appear in any of the LRA "Table 2's" for consistency with the GALL Report. If the enhancement listed in the SMP is credited to ensure that age-related degradation would be detected before a loss of intended function for the inaccessible concrete associated with further evaluation sections, then the accessible area line items need to be in the scope of the SMP and evaluated for consistency with GALL in Table 2's.

Request:

Provide a technical justification for why the following aging effects do not require management in accessible areas or identify a program to manage this aging effect. If a program is identified to manage this aging effect, update the LRA accordingly (including Table 2 AMR line items).

1. *increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide (SRP Table 3.5-1, Items 15 and 20)*
2. *cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel (SRP Table 3.5-1, Item 21)*
3. *increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack (SRP Table 3.5-1 Items 16 and 24)*

TVA Response to RAI 3.5.1-2

For each of the aging effects listed in the request, additional information is provided regarding whether the aging effect requires management.

1. LRA Table 3.5.1 (corrected number) Items 3.5.1-**15** and 3.5.1-**20** address the aging effect "Increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide and carbonation" for containment concrete components. Item 3.5.1-15 applies to containment component "Concrete (accessible areas): basemat." The SQN containment concrete is the circular concrete base foundation or basemat of the steel containment vessel (SCV) which is integral with the shield building concrete base foundation or basemat. The SCV concrete basemat is below the base liner plate of the SCV and, therefore, is not accessible. Because there is no accessible containment concrete, Item 3.5.1-15 was not referenced for SQN. Item 3.5.1-20 applies to containment component "Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (accessible areas): containment; wall; basemat." The NUREG-1801 items referencing this Item are associated with concrete containments and the SQN containment is a steel containment structure. Therefore, Item 3.5.1-20 was not applied for SQN. The changes to LRA Table 3.5.1 Items 3.5.1-15 and 3.5.1-20 are shown below.
2. LRA Table 3.5.1 Item 3.5.1-**21** addresses the aging effect "Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel" for containment concrete components. Item 3.5.1-21 applies to containment component "Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses; reinforcing steel, Concrete (accessible areas): basemat; reinforcing steel, Concrete (accessible areas): dome; wall; basemat; reinforcing steel." The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. The SCV concrete basemat is below the base liner plate of the SCV and, therefore, is not accessible. Because there is no accessible containment concrete, Item 3.5.1-21 was not referenced for SQN. The change to LRA Table 3.5.1 Item 3.5.1-21 is shown below.
3. LRA Table 3.5.1 Items 3.5.1-**16** and 3.5.1-**24** address the aging effect "Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack" for containment concrete components. Item 3.5.1-16 applies to containment component "Concrete (accessible areas): basemat, Concrete: containment; wall; basemat." The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. The SCV concrete basemat is below the base liner plate of the SCV and, therefore, is not accessible. Because there is no accessible containment concrete, Item 3.5.1-16 was not referenced for SQN. Item 3.5.1-24 applies to containment component "Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (inaccessible areas): basemat, Concrete (accessible areas): dome; wall; basemat." The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. Because the SCV

base foundation concrete is integral with the base foundation concrete of the shield building, the aging effect of the SCV base foundation concrete is managed along with the shield building base foundation concrete and is addressed in Table 3.5.1 Item 3.5.1-67 and LRA Table 3.5.2-1 line entry for component "Concrete (inaccessible areas): Shield building; below grade exterior; foundation." The Structures Monitoring Program manages the listed aging effect for the concrete (inaccessible areas) addressed by this line item. The changes to LRA Table 3.5.1 Item Numbers 3.5.1-16 and 3.5.1-24 are shown below.

4. TVA reviewed other Table 3.5.1 items not addressed in this RAI based on the staff's concern and evaluated them for consistency. As a result, TVA identified Table 3.5.1 Items 3.5.1-14, 3.5.1-18, 3.5.1-23, 3.5.1-25, 3.5.1-47, and 3.5.1-51 as needing clarification.

   LRA Table 3.5.1 Item 3.5.1-**14** addresses the aging effect "Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation" for containment concrete components. Item 3.5.1-14 applies to containment component "Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (inaccessible areas): containment; wall; basemat." The NUREG-1801 items referencing this Item are associated with concrete containments and SQN containment is a steel containment structure. Therefore, Item 3.5.1-14 was not applied for SQN. The changes to LRA Table 3.5.1 Item 3.5.1-14 and Section 3.5.2.2.1.9 are shown below.

   LRA Table 3.5.1 Item 3.5.1-**18** addresses the aging effect "Loss of material (spalling, scaling) and cracking due to freeze-thaw" for containment concrete components. Item 3.5.1-18 applies to containment component "Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (accessible areas): basemat." The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. The SCV concrete basemat is below the base liner plate of the SCV and therefore, is not accessible. Because there is no accessible containment concrete, Item 3.5.1-18 was not referenced for SQN. The change to LRA Table 3.5.1 Item Numbers 3.5.1-18 is shown below.

   LRA Table 3.5.1 Item 3.5.1-**23** addresses the aging effect "Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel" for containment concrete components. Item 3.5.1-23 applies to containment component "Concrete (inaccessible areas): basemat; reinforcing steel, Concrete (inaccessible areas): dome; wall; basemat; reinforcing steel." The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. Because the SCV base foundation concrete is integral with the base foundation concrete of the shield building, the aging effect of the SCV base foundation concrete is managed along with the shield building base foundation concrete and is addressed in Item 3.5.1-65 and LRA Table 3.5.2-1 line entry for component "Concrete (inaccessible areas): Shield building; below grade exterior; foundation." The Structures Monitoring Program manages the listed aging effect for the concrete (inaccessible areas) addressed by this line item. The change to LRA Table 3.5.1 Item 3.5.1-23 is shown below.

LRA Table 3.5.1 Item 3.5.1-**25** addresses the aging effect "Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel" for containment concrete components. Item 3.5.1-25 applies to containment component "Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses; reinforcing steel." The NUREG-1801 items referencing this Item are associated with PWR concrete containments and SQN containment is a steel containment. Therefore Item 3.5.1-25 was not applied for SQN. The change to LRA Table 3.5.1 Item 3.5.1-25 is shown below.

LRA Table 3.5.1 Items 3.5.1-**47** and 3.5.1-**51** address the aging effect "Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack" for non-containment concrete components. Item 3.5.1-47 applies to concrete component "Groups 1-5, 7-9: concrete (inaccessible areas): exterior above- and below-grade; foundation." Based on ongoing plant-specific operating experience (OE), increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in below-grade inaccessible concrete areas is an applicable aging effect for the SQN Groups 1-5 and 7-9 concrete structures and will be managed by the Structures Monitoring Program. Item 3.5.1-51 applies to concrete component "Group 6: concrete (inaccessible areas): exterior above- and below-grade; foundation; interior slab." Based on ongoing plant-specific OE, increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in below-grade inaccessible concrete areas is an applicable aging effect for the SQN Group 6 concrete structures and the Structures Monitoring Program will managed this aging effect. The changes to LRA Table 3.5.1 Items 3.5.1-47, 3.5.1-51, Sections 3.5.2.2.2.1 Item 4, 3.5.2.2.2.3 Item 3 and Tables 3.5.2-1, 3.5.2-2, 3.5.2-3 are shown below.

The changes to these LRA Sections and tables follow with additions underlined and deletions lined through:

LRA Sections 3.5.2.2.1.9, 3.5.2.2.2.1 Item 4, 3.5.2.2.2.3 Item 3, and

Table 3.5.1 Items 3.5.1-**14**, 3.5.1-**15**, 3.5.1-**16**, 3.5.1-**18**, 3.5.1-**20**, 3.5.1-**21**, 3.5.1-**23**, 3.5.1-**24**, 3.5.1-**25**, 3.5.1-**47**, 3.5.1-**51**, and

Tables 3.5.2-**1**, 3.5.2-**2**, and 3.5.2-**3**

"3.5.2.2.1.9 Increase in Porosity and Permeability due to Leaching of Calcium Hydroxide and Carbonation

The SQN containment is a low-leakage, free-standing SCV structure consisting of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. The SQN SCV base foundation is integral with the base foundation of the shield building.

The SQN SCV base foundation is designed in accordance with ACI 318-63 and constructed in accordance with the recommendations in ACI 318-63 and TVA's general construction specifications using ingredients/materials conforming to ACI and ASTM standards, which provide for a good quality, dense, well-cured, and low permeability concrete. Cracking is controlled through proper arrangement and distribution of reinforcing steel. The SQN SCV base foundation is constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. This is consistent with the recommendations and guidance provided by ACI 201.2R-77. Because the concrete base foundation is integral with the shield building concrete base foundation, it is not exposed to an environment conducive to this aging effect. Furthermore, ~~T~~the SQN SCV base foundation is not subject to the flowing water environment necessary for this aging effect to occur. Additionally, the SQN below-grade ground water environment is not aggressive (pH > 5.5, chlorides < 500 ppm, and sulfates < 1,500 ppm).

Therefore, increase in porosity and permeability due to leaching of calcium hydroxide and carbonation are not aging effects requiring management for the SQN SCV base foundation concrete.

3.5.2.2.2.1 Aging Management of Inaccessible Areas

4. Increase in Porosity and Permeability, and Loss of Strength due to Leaching of Calcium Hydroxide and Carbonation of Below-Grade Inaccessible Concrete Areas of Groups 1-5 and 7-9 Structures.

   The SQN Groups 1-5 and 7-9 concrete structures are designed in accordance with ACI 318-63 and ACI 318-71 and constructed in accordance with the recommendations in ACI 318-63, ACI 318-71 and TVA's general construction specifications using ingredients/materials conforming to ACI and ASTM standards, which provide for a good quality, dense, well-cured, and low permeability concrete. Cracking is controlled through proper arrangement and distribution of reinforcing steel. Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. This is consistent with the recommendations and guidance provided by ACI 201.2R-77. The SQN Groups 1-5 and 7-9 concrete structures are not subject to the flowing water environment necessary for this aging effect to occur. Additionally, the SQN below-grade ground water environment is not aggressive (pH > 5.5, chlorides < 500 ppm, and sulfates < 1,500 ppm). However, based on ongoing plant-specific operating experience, increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in below-grade inaccessible concrete areas is an applicable aging effect for the SQN Groups 1-5 and 7-9 concrete structures and is managed by the Structures Monitoring Program.

   ~~Therefore, increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in below-grade inaccessible concrete areas is not an applicable aging effect for the inaccessible concrete of SQN Groups 1-5 and 7-9 structures.~~

3.5.2.2.2.3 Aging Management of Inaccessible Areas for Group 6 Structures

For inaccessible areas of certain Group 6 structures, aging effects are covered by inspections in accordance with the Structures Monitoring program.

3. Increase in Porosity and Permeability and Loss of Strength due to Leaching of Calcium Hydroxide and Carbonation in Inaccessible Areas of Concrete Elements of Group 6 Structures

The SQN Group 6 concrete structures are designed in accordance with ACI 318-63 and ACI 318-71 and constructed in accordance with the recommendations in ACI 318-63, ACI 318-71, and TVA's general construction specifications using ingredients/materials conforming to ACI and ASTM standards, which provide for a good quality, dense, well-cured, and low permeability concrete. Cracking is controlled through proper arrangement and distribution of reinforcing steel. Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. This is consistent with the recommendations and guidance provided by ACI 201.2R-77. Additionally, the SQN below-grade ground water and raw water environments are not considered aggressive (pH > 5.5, chlorides < 500 ppm, and sulfates < 1,500 ppm). However, based on ongoing plant-specific operating experience, increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in below-grade inaccessible concrete areas is an applicable aging effect for the SQN Group 6 concrete structures and is managed by the Structures Monitoring Program.

~~Therefore, increase in porosity and permeability due to leaching of calcium hydroxide and carbonation is not an applicable aging effect requiring management for the inaccessible concrete of SQN Group 6 structures.~~"

Table 3.5.1: Structures and Component Supports					
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
PWR Concrete (Reinforced and Prestressed) and Steel Containments, BWR Concrete and Steel (Mark I, II, and III) Containments					
3.5.1-14	Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (inaccessible areas): containment; wall; basemat	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed.	Yes, if leaching is observed in accessible areas that impact intended function	~~Listed aging effects do not require management for the SQN concrete basemat.~~ NUREG-1801 items referencing this Item are associated with concrete containments and SQN containment is a steel containment. For further evaluation see Section 3.5.2.2.1.9.
3.5.1-15	Concrete (accessible areas): basemat	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	ISI (IWL).	No	~~Listed aging effects for the SQN SCV concrete basemat do not require management at SQN. SQN concrete is designed and constructed in accordance with ACI 318 with air-entrainment. Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. The design and construction of these structures at SQN prevents the effect of this aging from occurring; therefore, this aging effect does not require management. Aging effects are not significant for accessible areas. Nonetheless, the concrete basemat component is included in the Structures Monitoring Program to confirm the absence of these aging effects.~~ The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. The SCV concrete basemat is below the base liner plate of the SCV and, therefore, is not accessible. Because there is no accessible containment concrete, this Item is not referenced for SQN.

Table 3.5.1:	Structures and Component Supports				
Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-16	Concrete (accessible areas): basemat, Concrete: containment; wall; basemat	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	ISI (IWL) or Structures Monitoring Program	No	~~Listed aging effects for the SQN SCV concrete basemat do not require management at SQN. SQN concrete is designed and constructed in accordance with ACI 318 with air-entrainment. Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. The design and construction of these structures at SQN prevents the effect of this aging from occurring; therefore, this aging effect does not require management. Aging effects are not significant for accessible areas. Nonetheless, the concrete basemat component is included in the Structures Monitoring Program to confirm the absence of these aging effects.~~ The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. The SCV concrete basemat is below the base liner plate of the SCV and, therefore, is not accessible. Because there is no accessible containment concrete, this Item is not referenced for SQN.

Table 3.5.1: Structures and Component Supports					
Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-18	Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (accessible areas): basemat	Loss of material (spalling, scaling) and cracking due to freeze-thaw	ISI (IWL)	No	~~The SQN containment is a low-leakage free-standing SCV structure consisting of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. The SQN SCV base foundation is integral with the base foundation of the shield building. The base foundation of the SCV is below grade and protected from the outer environment by the shield building's base foundation and is not subject to freeze-thaw action. As a result, loss of material and cracking due to freeze-thaw are not aging effects requiring management for SQN SCV base foundation concrete. The absence of concrete aging effects for the SQN SCV base foundation concrete is confirmed under the Structures Monitoring Program.~~ The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. The SCV concrete basemat is below the base liner plate of the SCV and, therefore, is not accessible. Because there is no accessible containment concrete, this Item is not referenced for SQN.

Table 3.5.1: Structures and Component Supports					
Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-20	Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (accessible areas): containment; wall; basemat	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	ISI (IWL)	No	~~Listed aging effects for the SQN SCV concrete basemat do not require management at SQN. SQN concrete is designed and constructed in accordance with ACI 318 with air-entrainment. Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. The design and construction of these structures at SQN prevents the effect of this aging from occurring; therefore, this aging effect does not require management. Aging effects are not significant for accessible areas. Nonetheless, the concrete basemat component is included in the Structures Monitoring Program to confirm the absence of these aging effects.~~ NUREG-1801 items referencing this item are associated with concrete containments and SQN containment is a steel containment structure.

Table 3.5.1: Structures and Component Supports					
Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-21	Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses; reinforcing steel, Concrete (accessible areas): basemat; reinforcing steel, Concrete (accessible areas): dome; wall; basemat; reinforcing steel	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	ISI (IWL)	No	~~Listed aging effects for the SQN SCV concrete basemat do not require management at SQN. SQN concrete is designed and constructed in accordance with ACI 318 with air-entrainment. Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. The design and construction of these structures at SQN prevents the effect of this aging from occurring; therefore, this aging effect does not require management. Aging effects are not significant for accessible areas. Nonetheless, the concrete basemat component is included in the Structures Monitoring Program to confirm the absence of these aging effects.~~ The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. The SCV concrete basemat is below the base liner plate of the SCV and, therefore, is not accessible. Because there is no accessible containment concrete, this Item is not referenced for SQN.
3.5.1-23	Concrete (inaccessible areas): basemat; reinforcing steel, Concrete (inaccessible areas): dome; wall; basemat; reinforcing steel	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	ISI (IWL) or Structures Monitoring Program	No	~~Listed aging effects for the SQN SCV concrete basemat do not require management at SQN. SQN concrete is designed and constructed in accordance with ACI 318 with air-entrainment. Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. The design and construction of these structures at SQN prevents~~

Table 3.5.1: Structures and Component Supports					
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
					~~the effect of this aging from occurring; therefore, this aging effect does not require management. Aging effects are not significant for accessible and inaccessible areas. Nonetheless, the concrete basemat component is included in the Structures Monitoring Program to confirm the absence of these aging effects.~~ NUREG-1801 items referencing this Item are not associated with the SQN steel containment structure. The SQN steel containment structure has a circular concrete base foundation or basemat, which is integral with the shield building concrete base foundation or basemat. However, the aging effect for the concrete base foundation or basemat supporting the SCV structure is addressed in Item 3.5.1-65.
3.5.1-24	Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (inaccessible areas): basemat, Concrete (accessible areas): dome; wall; basemat	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	ISI (IWL) or Structures Monitoring Program	No	~~Listed aging effects for the SQN SCV concrete basemat do not require management at SQN. SQN concrete is designed and constructed in accordance with ACI 318 with air-entrainment. Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. The design and construction of these structures at SQN prevents the effect of this aging from occurring; therefore, this aging effect does not require management. Aging effects are not significant for accessible and inaccessible areas. Nonetheless, the concrete basemat component is included in the Structures Monitoring Program to confirm the absence of these aging effects.~~ NUREG-1801 items referencing this item are not associated with the SQN steel containment structure. The SQN SCV has a circular concrete

Table 3.5.1: Structures and Component Supports					
Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
					base foundation or basemat, which is integral with the shield building concrete base foundation or basemat. However, the aging effect for the concrete base foundation or basemat supporting the SQN SCV structure is addressed in Item 3.5.1-67.
3.5.1-25	Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses; reinforcing steel	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	ISI (IWL) or Structures Monitoring Program	No	~~Listed aging effects for the SQN SCV concrete basemat do not require management at SQN. SQN concrete is designed and constructed in accordance with ACI 318 with air-entrainment. Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. The design and construction of these structures at SQN prevents the effect of this aging from occurring; therefore, this aging effect does not require management. Aging effects are not significant for accessible and inaccessible areas. Nonetheless, the concrete basemat component is included in the Structures Monitoring Program to confirm the absence of these aging effects.~~ NUREG-1801 items referencing this item are associated with a concrete containment and SQN containment is a steel containment structure.
Safety-Related and Other Structures; and Component Supports					
3.5.1-47	Groups 1-5, 7-9: concrete (inaccessible areas): exterior above- and below-grade; foundation	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed.	Yes, if leaching is observed in accessible areas that impact intended function	~~Listed aging effects do not require management at SQN.~~ Consistent with NUREG-1801. The Structures Monitoring Program manages the listed aging effect. For further evaluation see Section 3.5.2.2.2.1 Item 4.

Table 3.5.1: Structures and Component Supports					
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
3.5.1-51	Groups 6: concrete (inaccessible areas): exterior above- and below-grade; foundation; interior slab	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed.	Yes, if leaching is observed in accessible areas that impact intended function	~~Listed aging effects do not require management at SQN.~~ Consistent with NUREG-1801. The Structures Monitoring Program manages the listed aging effect. For further evaluation see Section 3.5.2.2.2.3 Item 3.

Table 3.5.2-1: Reactor Building								
Structure and/or Component or Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Concrete (inaccessible areas): Shield building; below grade exterior; foundation	EN, FLB MB, PB, SNS, SRE, SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A1.TP-67	3.5.1-47	E

Table 3.5.2-2: Water Control Structures								
Structure and/or Component or Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Concrete (inaccessible areas): all	EN, FLB, HS, MB, SNS, SRE, SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A6.TP-109	3.5.1-51	E
Cable tunnel	MB, SRE	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A6.TP-109	3.5.1-51	E
Concrete cover for the rock walls of approach channel	EN, SNS	Concrete	Exposed to fluid environment	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A6.TP-109	3.5.1-51	E
Discharge box and foundation	EN, MB, SRE, SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A6.TP-109	3.5.1-51	E
Exterior concrete slabs and concrete caps	MB, SRE	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A6.TP-109	3.5.1-51	E

Table 3.5.2-3: Turbine Building, Aux/Control Building and Other Structures								
Structure and/or Component or Commodity	**Intended Function**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801 Item**	**Table 1 Item**	**Notes**
Concrete (inaccessible areas); below-grade exterior; foundation	EN, FLB, MB, PB, SNS, SRE, SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A3.TP-67	3.5.1-47	E
Cable tunnel	MB, SRE	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A3.TP-67	3.5.1-47	E
Concrete slab (missile barrier)	MB	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A7.TP-67	3.5.1-47	E
Duct banks	EN, SNS, SRE, SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A3.TP-67	3.5.1-47	E
Foundations (e.g., switchyard, transformers, tanks, circuit breakers)	SNS, SRE, SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A3.TP-67	3.5.1-47	E
Manholes and handholes	EN, SNS, SRE, SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A3.TP-67	3.5.1-47	E
Pipe tunnel	MB, PB, SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A3.TP-67	3.5.1-47	E
RWST storage basin	SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A3.TP-67	3.5.1-47	E
Sumps	SNS, SRE, SSR	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A3.TP-67	3.5.1-47	E
Trenches	EN, SNS	Concrete	Soil	Increase in porosity and permeability; loss of strength	Structures Monitoring	III.A3.TP-67	3.5.1-47	E

RAI 3.5.1-1a

Background:

LRA Table 3.5-1 (sic, 3.5.1), items 3.5.1-12 and 3.5.1-19 address cracking due to expansion from reaction with aggregates in inaccessible and accessible areas of containment concrete; respectively. The applicant's response to RAI 3.5-1 indicated that it would manage this aging effect, for areas of accessible and inaccessible concrete associated with LRA Table 3.5-1, Items 43, 50, and 54, using the Structures Monitoring Program.

Issue:

The staff noted that items 3.5.1-12 and 3.5.1-19 were not included in RAI 3.5.1-1; however, they also address cracking due to expansion from reaction with aggregates. As stated in RAI 3.5.1-1, regardless of the design and construction of the concrete, the staff believes all aging effects could occur in accessible areas and therefore, require management. The discussion in the LRA states that the components are included in the SMP; however, the associated line items do not appear in any of the LRA "Table 2's."

Request:

State whether LRA Table 3.5-1 items 3.5.1-12 and 3.5.1-19 will be revised consistent with those revised in response to RAI 3.5.1-1. If a program is identified to manage this aging effect, update the LRA accordingly. If not, provide a technical justification for why cracking due to reaction with aggregates does not require management in accessible or inaccessible areas of the concrete basemat.

TVA Response to RAI 3.5.1-1a

The discussion of LRA Table 3.5.1 (corrected number) items 3.5.1-12 and 3.5.1-19 below includes clarification regarding how the effects of aging are managed.

- Item 3.5.1-12 discusses the aging effect "Cracking due to expansion from reaction with aggregates" for component "Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (inaccessible areas): basemat, Concrete (inaccessible areas): containment; wall; basemat, Concrete (inaccessible areas): basemat, concrete fill-in annulus." The inaccessible containment concrete associated with this item is the circular concrete base foundation or basemat supporting the SCV. The containment concrete foundation is integral with the concrete foundation of the shield building housing the SCV, therefore, the Structures Monitoring Program (SMP) manages the effects of aging for the inaccessible containment concrete along with the concrete foundation of the shield building. The applicable component in LRA Table 3.5.2-1 is "Concrete (inaccessible areas): Shield building; below grade exterior; foundation", which references LRA Table 3.5.1, item 3.5.1-43 as shown in the response to RAI 3.5.1-1 (ADAMS No. ML13213A026). The changes to LRA Section 3.5.2.2.1.8 and Table 3.5.1 item 3.5.1-12 are shown below.
- Item 3.5.1-19 discusses the aging effect "Cracking due to expansion from reaction with aggregates" for component "Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (accessible areas): basemat, Concrete (accessible areas)

containment; wall; basemat, concrete fill-in annulus." The containment concrete associated with item 3.5.1-19 is the circular concrete base foundation or basemat of the SQN SCV. The concrete basemat is below the base liner plate of the SCV and, therefore, is not accessible. Because there is no accessible containment concrete, this item number is not applicable for SQN. The change to LRA Table 3.5.1 item 3.5.1-19 is shown below.

The changes to LRA Section 3.5.2.2.1.8 and Table 3.5.1 items 3.5.1-12 and 3.5.1-19 follow with additions underlined and deletions lined through.

"3.5.2.2.1.8 Cracking due to Expansion from Reaction with Aggregate

The SQN containment is a low-leakage, free-standing SCV structure consisting of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. The SQN SCV base foundation is integral with the base foundation of the shield building.

The SQN SCV base foundation is designed in accordance with ACI 318-63 and constructed in accordance with the recommendations in ACI 318-63 and TVA's general construction specifications using ingredients/materials conforming to ACI and ASTM standards. The concrete mix uses Portland cement conforming to ASTM C150, Type II along with fly ash (ASTM C618, Class F). Concrete aggregates conform to the requirements of ASTM C33. The aggregate used in the concrete of the SQN components did not come from a region known to yield aggregates suspected of or known to cause aggregate reactions. Materials for concrete used in SQN structures and components were specifically investigated, tested, and examined in accordance with pertinent ASTM standards. All aggregates used at SQN conform to the requirements of ASTM C33, "Standard Specification of Concrete Aggregates." Appendix X1 of ASTM C33 identifies methods for evaluating potential reactivity of aggregates, including ASTM C295, ASTM C289, ASTM C227, and ASTM C342. Also, use of a low alkali Portland cement (ASTM C150 Type II) containing less than 0.60 percent alkali calculated as sodium oxide equivalent was required by TVA's general construction specifications and will prevent harmful expansion due to alkali aggregate reaction. Additionally, water/cement ratios were within the limits provided in ACI 318. <u>Based on ongoing industry operating experience, cracking due to expansion from reaction with aggregate in below-grade inaccessible concrete areas is considered an applicable aging effect for the containment base foundation concrete. Because the SQN SCV base foundation concrete is integral with the base foundation concrete of the shield building, the Structures Monitoring Program manages the effects of aging on the SCV base foundation concrete along with the shield building base foundation concrete.</u>

~~Therefore, cracking due to expansion from reaction with aggregate is not an aging effect requiring management for the SQN SCV base foundation concrete. "~~

Aging Management of Inaccessible Areas for Group 6 Structures

Table 3.5.1: Structures and Component Supports					
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
PWR Concrete (Reinforced and Prestressed) and Steel Containments, BWR Concrete and Steel (Mark I, II, and III) Containments					
3.5.1-12	Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (inaccessible areas): basemat, Concrete (inaccessible areas): containment; wall; basemat, Concrete (inaccessible areas): basemat, concrete fill-in annulus	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed.	Yes, if concrete is not constructed as stated function	~~Listed aging effects do not require management for the SQN concrete basemat.~~ The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. However, the aging effect for the concrete base foundation or basemat is addressed in Item 3.5.1-43. For further evaluation see Section 3.5.2.2.1.8.

Table 3.5.1: Structures and Component Supports					
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
3.5.1-19	Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses, Concrete (accessible areas): basemat, Concrete (accessible areas) containment; wall; basemat, concrete fill-in annulus	Cracking due to expansion from reaction with aggregates	ISI (IWL)	No	~~Listed aging effects for the SQN SCV concrete basemat do not require management at SQN. SQN concrete is designed and constructed in accordance with ACI 318 with air-entrainment. Concrete aggregates conform to the requirements of ASTM C33. The aggregate used in the concrete of the SQN components did not come from a region known to yield aggregates suspected of or known to cause aggregate reactions. The design and construction of these structures at SQN prevents the effect of this aging from occurring; therefore, this aging effect does not require management. Aging effects are not significant for accessible areas. Nonetheless, the concrete basemat component is included in the Structures Monitoring Program to confirm the absence of these aging effects.~~ <ins>The SQN containment concrete is the circular concrete base foundation or basemat of the SCV which is integral with the shield building concrete base foundation or basemat. The SCV concrete basemat is below the base liner plate of the SCV and, therefore, is not accessible. Because there is no accessible containment concrete, this Item is not referenced for SQN.</ins>

RAI B.1.6-1a

<u>Background:</u>

In its response to item 1 of RAI B.1.6-1, on July 1, 2013, the applicant provided an Exhibit A showing the design modification for the test connection tubing in the access boxes installed in SQN Unit 2, and stated that plans are in place to install a similar modification in SQN Unit 1. The applicant also stated "prior to installing this design modification in SQN Unit 2, remote visual examinations were performed, to the extent possible, inside the leak test channels by inserting a boroscope video probe into test connection tubing. Based on the satisfactory examination results to date, following installation of the design modification SQN has no plans to perform future visual examinations of the embedded SCV liner plate or embedded leak test channels."

GALL Report AMP XI.S1, program element "detection of aging effects," states "[t]he examination methods, frequency, and scope of examination specified in 10 CFR 50.55a and Subsection IWE ensure that aging effects are detected before they compromise the design-basis requirements." 10 CFR 50.55a(b)(2)(ix)(A) states that licensees "shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could be indicative of or result in degradation of inaccessible areas."

<u>Issue:</u>

Exhibit A shows a cover plate seal welded to the bottom of the corroded access boxes. The all-around field welding symbol pointing to the welding of the cover plates to access box steels, may not meet code approved welding standards because of their degraded (excessive corrosion) condition. The SQN-NRC Integrated Inspection Report (IR)-2012005 of February 13, 2013, states that an inspection completed by NRC on December 31, 2012 indicated the failure of the applicant to conduct IWE visual inspections of the access boxes. Furthermore, the IR states that the applicant subsequently performed visual examinations that revealed significant corrosion of the access boxes, including a through-wall hole in tubing leading down to a leak chase channel. Follow-up boroscopic examination confirmed the existence of water in the leak chase channels with corrosion.

It is not clear whether the applicant's design modification to cover the tubing opening is an effective approach of sealing the leak channel test connection. It is also not clear why "SQN has no plans to perform future visual examinations of the embedded SCV liner plate or embedded leak test channels."

<u>Request:</u>

1. *Explain how the design modification, shown in exhibit A, will be effective in sealing the leak chase channels from moisture intrusion during the period of extended operation. Furthermore complete exhibit A, shown in RAI B.1.6-1, with a code approved weld type, weld-size, and weld symbol continued to be used for welding the cover plates to the access box steels.*

2. *Explain why the applicant has no plans to perform future visual examinations of the embedded leak test channels, when the recent IR indicates the existence of water in the channels and corresponding corrosion.*

TVA Response to RAI B.1.6-1a

1. At SQN, visual inspection in 2012 identified standing water and corrosion on the inside surfaces of the ¼-inch thick carbon steel access boxes (Item 1 of Exhibit A) and the base slab floor penetrations (Item 3 of Exhibit A). The more significant corrosion, including through-wall corrosion at several locations, was identified on the thin-walled ¾-inch diameter carbon steel pressure test connection tubing (Item 4 of Exhibit A) beneath the bottom plate of the access box. This location was most susceptible to corrosion because any water that bypassed the access box lid gasket drained into the annular area between the floor penetration pipe (Item 3 of Exhibit A) and the test connection tubing (Item 4 of Exhibit A), where it could be in contact with the outside surface of the ¾-inch diameter test connection tubing. Based on these findings, the original design configuration of SQN Unit 2 was modified so that the test connection tubing no longer served the function of preventing moisture from entering the pressure test channel (Item 5 of Exhibit A). The corroded test connection tubing was cut off below the access box (Item 1 of Exhibit A) and abandoned in place. A ¼-inch carbon steel plate (Item 2 of Exhibit A) was welded over the floor penetration pipe (Item 3 of Exhibit A). A non-structural seal weld was applied, as depicted in Exhibit A, to eliminate possible moisture intrusion into the test connection tubing and the floor penetration. As required by SQN welding procedures, the surfaces were cleaned by grinding to base metal and visually inspected prior to welding.

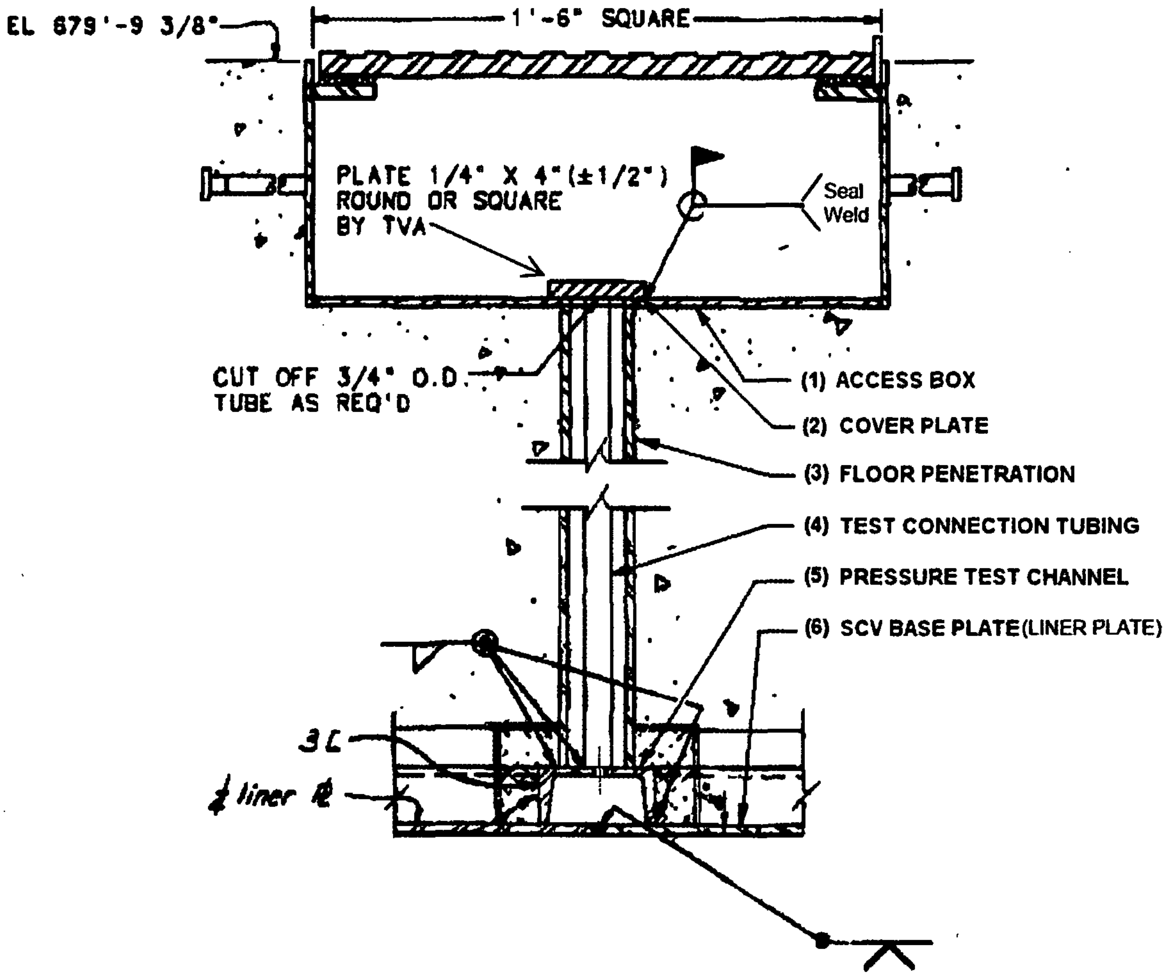

EXHIBIT A: TYPICAL TEST CONNECTION DETAIL

2. When water and corrosion were identified in the access boxes in 2012, inspections were performed on the embedded SCV base liner plate surface by inserting a boroscope inside the test connection tubing, through the pressure test channel openings, in compliance with 10 CFR 50.55a(b)(2)(ix)(A). TVA discussed in the response to RAI B.1.6-1 (ADAMS No. ML13190A276), SQN modified the access box configuration in Unit 2 to prevent water intrusion into the annular area between the 2-inch diameter pipe sleeve (Item 3 of Exhibit A) and the ¾-inch diameter test connection tubing (Item 4 of Exhibit A) and into the pressure test channel (Item 5 of Exhibit A). The modification described in response to Item 1 above prevents water that enters the access box from draining into the floor penetration piping, thus preventing moisture from contacting the test connection tubing, the pressure test channel, and the SCV liner plate surfaces. Because this modification provides a robust water intrusion barrier, the need for SQN to routinely remove the welded cover plates (Item 2 of Exhibit A) to access the embedded portions of the SCV liner plate is not necessary. Once the modifications are

complete for Unit 1, water, if any, that penetrates the access box gasket will be captured within the access box. There is no viable flowpath unless a through-wall flaw occurs in the access box base metal, cover plate, or weld. However, if conditions are identified for the accessible areas that would indicate potential degradation of inaccessible areas, further visual examinations beneath the welded cover plates will be performed in accordance with 10 CFR 50.55a(b)(2)(ix)(A). To monitor the condition of the access boxes and associated materials, SQN has implemented an examination program for inspections of the access boxes. Visual examinations of all accessible surfaces, including the access box surfaces, cover plate, welds, and gasket sealing surfaces are performed at the access boxes on each unit every other refueling outage with the gasketed access box lid removed.

RAI B.1.6-1b

Background:

In its response to item 2 of RAI B.1.6-1 on July 1, 2013, the applicant stated "[b]ased on past satisfactory examinations results, SQN has no plans to perform ultrasonic tests (UT) examination of the SCV below the moisture barrier from the annulus area or from inside the SCV." The applicant also stated that "if future examinations identify moisture intrusion below the moisture barrier sealant in the inaccessible area of SCV embedded in concrete, one or both of these examination techniques may be necessary for compliance with 10 CFR 50.55a(b)(2)(ix), and would be performed if necessary."

Issue:

It is not clear what examination techniques the applicant is referring to use if moisture intrusion below the moisture barrier sealant in the inaccessible area of SCV embedded in concrete were identified during the period of extended operation.

Request:

Identify what examination techniques are to be used, if moisture intrusion below the moisture barrier sealant in the inaccessible area of SCV embedded in concrete were identified during the period of extended operation.

TVA Response to RAI B.1.6-1b

If moisture intrusion is identified below the moisture barrier sealant in the inaccessible area of the SCV embedded in concrete during the PEO, SQN would perform visual examination, ultrasonic testing (UT), or other proven non-destructive examination techniques on the SCV as necessary to determine the extent of wall loss and comply with 10 CFR 50.55a(b)(2)(ix).

RAI B.1.6-2a

Background:

In its response to item 1 of RAI B.1.6-2 on July 1, 2013, the applicant stated "SQN elected to perform augmented volumetric examinations at the location of the full penetration welds where the SCV domes were cut for the steam generator replacements (SGRs). This voluntary volumetric examination is not required by the ASME Code and change to this examination does not represent a change in scope to the requirements established under IWE-2412. IWE-2412 is not applicable to the examination frequency for this owner elected examination."

In its response to item 2 of RAI B.1.6-2 on July 1, 2013, the applicant stated "A similar owner-elected augmented examination plan was performed at Tennessee Valley Authority Watts Bar Nuclear Plant. The volumetric examinations are strictly voluntary examinations beyond those required by the ASME Code and do not constitute a change in scope to the requirements established under IWE-2412."

The staff noted, however, the following ASME Section XI, IWE and referenced Articles:

- *IWE-1241 "Examination Surface Areas," that states "Surface areas likely to experience accelerated degradation and aging require the augmented examinations identified in Table IWE-2500-1, Examination Category E-C."*
- *IWE-2500(b)(4) "Examination and Pressure Test Requirements," which states that "... periodic reexamination can be performed in accordance with the requirements of Table IWE-2500-1, Examination Category E-C."*

In addition the staff noted in the GALL Report, XI.S2, ASME Section XI, Subsection IWE program description that "[l]imited volumetric examination (ultrasonic thickness measurement) and surface examination (e.g., liquid penetrant) may also be necessary in some instances to detect aging effects." Specifically:

- *"Scope of program," program element, states "The components within the scope of Subsection IWE are Class MC pressure-retaining components (steel containments) and their integral attachments, metallic shell and penetration liners of Class CC containments and their integral attachments, containment moisture barriers, containment pressure-retaining bolting, and metal containment surface areas, including welds and base metal;" and*
- *"Detection of aging effects," program element, states "IWE-1240 requires augmented examinations (Examination Category E-C) of containment surface areas subject to degradation. A VT-1 visual examination is performed for areas accessible from both sides, and volumetric (ultrasonic thickness measurement) examination is performed for areas accessible from only one side."*

Issue:

1. *The staff reviewed the applicant's response and noted that it identifies volumetric examination at the locations of the full penetration welds where the SCV domes were*

cut, as voluntary and not required by ASME Code of record. The applicant also stated that "changes to this examination do not represent a change in scope to the requirements established under IWE-2412. IWE-2412 is not applicable to the examination frequency for this owner-elected examination."

a. *It is not clear whether the surface areas of the SCV subject to volumetric examinations are experiencing accelerated degradation, requiring ultrasonic thickness examination per IWE-1241 augmented examination, as listed in Examination Category of E-C of Table IWE2500-1; and*

b. *It is not clear why IWE-2412 is not applicable to the examination frequency for the owner-elected examination.*

2. *Furthermore, the applicant did not provide any discussion(s) on fleet-wide operating experience(s) and associated corrective actions that may have been performed, and are the cause of applicant's "voluntary" volumetric examinations at the locations of the full penetration welds where the SCV domes were cut.*

Request:

1. *Explain whether:*

   a. *the augmented volumetric examinations are pursued because of anticipated aging effects experiencing accelerated degradation at the locations of the full penetration welds where the SCV domes were cut; and*

   b. *the IWE-2412 examination frequency will continue to be performed during the period of extended operation.*

2. *Provide operating experience(s) and associated corrective action(s) for any past volumetric examination(s) performed to ensure the integrity of the SCVs continue to be maintained across the fleet.*

TVA Response to RAI B.1.6-2a

1.a. The owner-elected volumetric examinations are performed at the locations of the full penetration welds where the SCV domes were cut and coatings on the inside of SCV were not reinstalled following steam generator replacement. The examinations are not being performed because of anticipated aging effects causing accelerated degradation. These locations, on the underside of the containment dome, are exposed only to an air-indoor uncontrolled in the containment atmosphere. This owner-elected examination is not an ASME Code augmented examination: therefore, it is not being performed in accordance with ASME Code Section XI, Examination Category E-C of Table IWE-2500-1. These examinations were not the result of industry operating experience with accelerated corrosion at this location.

1.b. As discussed in Response 1.a, the volumetric examination is solely an owner-elected examination and is not an examination required by ASME Code Section XI. Although the examinations are performed at the Article IWE-2412 examination frequency, the ASME Code is not the basis for this examination and the examination frequency may be modified during the

PEO. Examinations will continue at the frequency determined by SQN engineering until the coatings are reinstalled.

2. There is no known fleet or industry OE with accelerated corrosion at this location, and no associated corrective actions to report. Additionally, the "OE" discussed in the response to RAI B.1.6-2 (ADAMS No. ML13190A276) refers to the TVA fleet's trend of essentially unchanged thickness measurements since inception of the UT examination in 2003 as the basis for revising the owner-elected examination frequency.

ENCLOSURE 3

Tennessee Valley Authority
Sequoyah Nuclear Plant, Units 1 and 2 License Renewal

Regulatory Commitment List, Revision 8

Commitment **7.D** has been revised. Additions are underlined.

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
1	Implement the **Aboveground Metallic Tanks** Program as described in LRA Section B.1.1	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.1
2	A. Revise **Bolting Integrity Program** procedures to ensure the actual yield strength of replacement or newly procured bolts will be less than 150 ksi B. Revise Bolting Integrity Program procedures to include the additional guidance and recommendations of EPRI NP-5769 for replacement of ASME pressure-retaining bolts and the guidance provided in EPRI TR-104213 for the replacement of other pressure-retaining bolts. C. Revise Bolting Integrity Program procedures to specify a corrosion inspection and a check-off for the transfer tube isolation valve flange bolts. D. Revise Bolting Integrity Program procedures to visually inspect a representative sample of normally submerged ERCW system bolts at least once every 5 years. (See Set 10 (30-day), Enclosure 1, B.1.2-2a)	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.2
3	A. Implement the **Buried and Underground Piping and Tanks Inspection Program** as described in LRA Section B.1.4. B. Cathodic protection will be provided based on the guidance of NUREG-1801, section XI.M41, as modified by LR-ISG-2011-03.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.4

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
4	A. Revise **Compressed Air Monitoring Program** procedures to include the standby diesel generator (DG) starting air subsystem. B. Revise Compressed Air Monitoring Program procedures to include maintaining moisture and other contaminants below specified limits in the standby DG starting air subsystem. C. Revise Compressed Air Monitoring Program procedures to apply a consideration of the guidance of ASME OM-S/G-1998, Part 17; EPRI NP-7079; and EPRI TR-108147 to the limits specified for the air system contaminants D. Revise Compressed Air Monitoring Program procedures to maintain moisture, particulate size, and particulate quantity below acceptable limits in the standby DG starting air subsystem to mitigate loss of material. E. Revise Compressed Air Monitoring Program procedures to include periodic and opportunistic visual inspections of surface conditions consistent with frequencies described in ASME O/M-SG-1998, Part 17 of accessible internal surfaces such as compressors, dryers, after-coolers, and filter boxes of the following compressed air systems: • Diesel starting air subsystem • Auxiliary controlled air subsystem • Nonsafety-related controlled air subsystem F. Revise Compressed Air Monitoring Program procedures to monitor and trend moisture content in the standby DG starting air subsystem. G. Revise Compressed Air Monitoring Program procedures to include consideration of the guidance for acceptance criteria in ASME OM-S/G-1998, Part 17, EPRI NP-7079; and EPRI TR-108147.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.5

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
5	A. Revise **Diesel Fuel Monitoring Program** procedures to monitor and trend sediment and particulates in the standby DG day tanks. B. Revise Diesel Fuel Monitoring Program procedures to monitor and trend levels of microbiological organisms in the seven-day storage tanks. C. Revise Diesel Fuel Monitoring Program procedures to include a ten-year periodic cleaning and internal visual inspection of the standby DG diesel fuel oil day tanks and high pressure fire protection (HPFP) diesel fuel oil storage tank. These cleanings and internal inspections will be performed at least once during the ten-year period prior to the period of extended operation and at succeeding ten-year intervals. If visual inspection is not possible, a volumetric inspection will be performed. D. Revise Diesel Fuel Monitoring Program procedures to include a volumetric examination of affected areas of the diesel fuel oil tanks, if evidence of degradation is observed during visual inspection. The scope of this enhancement includes the standby DG seven-day fuel oil storage tanks, standby DG fuel oil day tanks, and HPFP diesel fuel oil storage tank and is applicable to the inspections performed during the ten-year period prior to the period of extended operation and succeeding ten-year intervals.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.8
6	A. Revise **External Surfaces Monitoring Program** procedures to clarify that periodic inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3) will be performed. Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2). B. Revise External Surfaces Monitoring Program procedures to include instructions to look for the following related to metallic components: • Corrosion and material wastage (loss of material). • Leakage from or onto external surfaces loss of material). • Worn, flaking, or oxide-coated surfaces (loss of material). • Corrosion stains on thermal insulation (loss of material). • Protective coating degradation (cracking, flaking, and blistering). • Leakage for detection of cracks on the external surfaces of stainless steel components exposed to an air environment containing halides. C. Revise External Surfaces Monitoring Program procedures to include instructions for monitoring aging effects for flexible polymeric components, including manual or physical manipulations of the material, with a sample size for manipulation of at least ten	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.10

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(6)	percent of the available surface area. The inspection parameters for polymers shall include the following: • Surface cracking, crazing, scuffing, dimensional changes (e.g., ballooning and necking) -). • Discoloration. • Exposure of internal reinforcement for reinforced elastomers (loss of material). • Hardening as evidenced by loss of suppleness during manipulation where the component and material can be manipulated. D. Revise External Surfaces Monitoring Program procedures to ensure surfaces that are insulated will be inspected when the external surface is exposed (i.e., during maintenance) at such intervals that would ensure that the components' intended function is maintained. E. Revise External Surfaces Monitoring Program procedures to include acceptance criteria. Examples include the following: • Stainless steel should have a clean shiny surface with no discoloration. • Other metals should not have any abnormal surface indications. • Flexible polymers should have a uniform surface texture and color with no cracks and no unanticipated dimensional change, no abnormal surface with the material in an as-new condition with respect to hardness, flexibility, physical dimensions, and color. • Rigid polymers should have no erosion, cracking, checking or chalks.		

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
7	A. Revise **Fatigue Monitoring Program** procedures to monitor and track critical thermal and pressure transients for components that have been identified to have a fatigue Time Limited Aging Analysis. B. Fatigue usage calculations that consider the effects of the reactor water environment will be developed for a set of sample reactor coolant system (RCS) components. This sample set will include the locations identified in NUREG/CR-6260 and additional plant-specific component locations in the reactor coolant pressure boundary if they are found to be more limiting than those considered in NUREG/CR-6260. In addition, fatigue usage calculations for reactor vessel internals (lower core plate and control rod drive (CRD) guide tube pins) will be evaluated for the effects of the reactor water environment. F_{en} factors will be determined as described in Section 4.3.3. C. Fatigue usage factors for the RCS pressure boundary components will be adjusted as necessary-to incorporate the effects of the Cold Overpressure Mitigation System (COMS) event (i.e., low temperature overpressurization event) and the effects of structural weld overlays. D. Revise Fatigue Monitoring Program procedures to provide updates of the fatigue usage calculations <u>and cycle-based fatigue waiver evaluations</u> on an as-needed basis if an allowable cycle limit is approached, or in a case where a transient definition has been changed, unanticipated new thermal events are discovered, or the geometry of components have been modified. E. Revise Fatigue Monitoring Program procedures to track the tensioning cycles for the reactor coolant pump hydraulic studs.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.11
8	A. Revise **Fire Protection Program** procedures to include an inspection of fire barrier walls, ceilings, and floors for any signs of degradation such as cracking, spalling, or loss of material caused by freeze thaw, chemical attack, or reaction with aggregates. B. Revise Fire Protection Program procedures to provide acceptance criteria of no significant indications of concrete cracking, spalling, and loss of material of fire barrier walls, ceilings, and floors and in other fire barrier materials.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.12
9	A. Revise Fire Water System Program procedures to include periodic visual inspection of fire water system internals for evidence of corrosion and loss of wall thickness. B. Revise Fire Water System Program procedures to include one of the following options: • Wall thickness evaluations of fire protection piping using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material will be performed prior to the period of	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.13

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(9)	extended operation and periodically thereafter. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function. • A visual inspection of the internal surface of fire protection piping will be performed upon each entry into the system for routine or corrective maintenance. These inspections will be capable of evaluating (1) wall thickness to ensure against catastrophic failure and (2) the inner diameter of the piping as it applies to the design flow of the fire protection system. Maintenance history shall be used to demonstrate that such inspections have been performed on a representative number of locations prior to the period of extended operation. A representative number is 20% of the population (defined as locations having the same material, environment, and aging effect combination) with a maximum of 25 locations. Additional inspections will be performed as needed to obtain this representative sample prior to the period of extended operation and periodically during the period of extended operation based on the findings from the inspections performed prior to the period of extended operation. C. Revise Fire Water System Program procedures to ensure a representative sample of sprinkler heads will be tested or replaced before the end of the 50-year sprinkler head service life and at ten-year intervals thereafter during the extended period of operation. NFPA-25 defines a representative sample of sprinklers to consist of a minimum of not less than four sprinklers or one percent of the number of sprinklers per individual sprinkler sample, whichever is greater. If the option to replace the sprinklers is chosen, all sprinkler heads that have been in service for 50 years will be replaced. D. Revise the Fire Water System Program full flow testing to be in accordance with full flow testing standards of NFPA-25 (2011). E. Revise Fire Water System Program procedures to include acceptance criteria for periodic visual inspection of fire water system internals for corrosion, minimum wall thickness, and the absence of biofouling in the sprinkler system that could cause corrosion in the sprinklers.		
10	A. Revise **Flow Accelerated Corrosion** (FAC) **Program** procedures to implement NSAC-202L guidance for examination of components upstream of piping surfaces where significant wear is detected. B. Revise FAC Program procedures to implement the guidance in LR-ISG-2012-01, which will include a susceptibility review based on internal operating experience, external operating experience, EPRI TR-1011231, *Recommendations for Controlling Cavitation, Flashing, Liquid Droplet Impingement, and Solid Particle Erosion in Nuclear Power Plant Piping*, and NUREG/CR-6031, *Cavitation Guide for Control Valves*.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.14

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
11	Revise **Flux Thimble Tube Inspection Program** procedures to include a requirement to address if the predictive trending projects that a tube will exceed 80% wall wear prior to the next planned inspection, then initiate a Service Request (SR) to define actions (i.e., plugging, repositioning, replacement, evaluations, etc.) required to ensure that the projected wall wear does not exceed 80%. If any tube is found to be >80% through wall wear, then initiate a Service Request (SR) to evaluate the predictive methodology used and modify as required to define corrective actions (i.e., plugging, repositioning, replacement, etc).	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.15
12	A. Revise **Inservice Inspection–IWF Program** procedures to clarify that detection of aging effects will include monitoring anchor bolts for loss of material, loose or missing nuts, and cracking of concrete around the anchor bolts. B. Revise ISI - IWF Program procedures to include the following corrective action guidance. When a component support is found with minor age-related degradation, but still is evaluated as "acceptable for continued service" as defined in IWF-3400, the program owner may choose to repair the degraded component. If the component is repaired, the program owner will substitute a randomly selected component that is more representative of the general population for subsequent inspections.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.17
13	Inspection of **Overhead Heavy Load and Light Load** (Related to Refueling) **Handling Systems**: A. Revise program procedures to specify the inspection scope will include monitoring of rails in the rail system for wear; monitoring structural components of the bridge, trolley and hoists for the aging effect of deformation, cracking, and loss of material due to corrosion; and monitoring structural connections/bolting for loose or missing bolts, nuts, pins or rivets and any other conditions indicative of loss of bolting integrity. B. Revise program procedures to include the inspection and inspection frequency requirements of ASME B30.2. C. Revise program procedures to clarify that the acceptance criteria will include requirements for evaluation in accordance with ASME B30.2 of significant loss of material for structural components and structural bolts and significant wear of rail in the rail system. D. Revise program procedures to clarify that the acceptance criteria and maintenance and repair activities use the guidance provided in ASME B30.2	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.18
14	Implement the **Internal Surfaces in Miscellaneous Piping and Ducting Components Program** as described in LRA Section B.1.19.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.19

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
15	Implement the **Metal Enclosed Bus Inspection Program** as described in LRA Section B.1.21.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.21
16	A. Revise **Neutron Absorbing Material Monitoring Program** procedures to perform blackness testing of the Boral coupons within the ten years prior to the period of extended operation and at least every ten years thereafter based on initial testing to determine possible changes in boron-10 areal density. B. Revise Neutron Absorbing Material Monitoring Program procedures to relate physical measurements of Boral coupons to the need to perform additional testing. C. Revise Neutron Absorbing Material Monitoring Program procedures to perform trending of coupon testing results to determine the rate of degradation and to take action as needed to maintain the intended function of the Boral.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.22
17	Implement the **Non-EQ Cable Connections Program** as described in LRA Section B.1.24	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.24
18	Implement the **Non-EQ Inaccessible Power Cable** (400 V to 35 kV) **Program** as described in LRA Section B.1.25	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.25
19	Implement the **Non-EQ Instrumentation Circuits Test Review Program** as described in LRA Section B.1.26.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.26
20	Implement the **Non-EQ Insulated Cables and Connections Program** as described in LRA Section B.1.27	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.27
21	A. Revise **Oil Analysis Program** procedures to monitor and maintain contaminants in the 161-kV oil filled cable system within acceptable limits through periodic sampling in accordance with industry standards, manufacturer's recommendations and plant-specific operating experience. B. Revise Oil Analysis Program procedures to trend oil contaminant levels and initiate a problem evaluation report if contaminants exceed alert levels or limits in the 161-kV oil-filled cable system.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.28
22	Implement the **One-Time Inspection Program** as described in LRA Section B.1.29.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.29
23	Implement the **One-Time Inspection – Small Bore Piping Program** as described in LRA Section B.1.30	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.30
24	Revise **Periodic Surveillance and Preventive Maintenance Program** procedures as necessary to include all activities described in the table provided in the LRA Section B.1.31 program description.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.31

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
25	A. Revise **Protective Coating Program** procedures to clarify that detection of aging effects will include inspection of coatings near sumps or screens associated with the emergency core cooling system. B. Revise Protective Coating Program procedures to clarify that instruments and equipment needed for inspection may include, but not be limited to, flashlights, spotlights, marker pen, mirror, measuring tape, magnifier, binoculars, camera with or without wide-angle lens, and self-sealing polyethylene sample bags. C. Revise Protective Coating Program procedures to clarify that the last two performance monitoring reports pertaining to the coating systems will be reviewed prior to the inspection or monitoring process.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.32
26	A. Revise **Reactor Head Closure Studs Program** procedures to ensure that replacement studs are fabricated from bolting material with actual measured yield strength less than 150 ksi. B. Revise Reactor Head Closure Studs Program procedures to exclude the use of molybdenum disulfide (MoS_2) on the reactor vessel closure studs and to refer to Reg. Guide 1.65, Rev1.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.33
27	A. Revise **Reactor Vessel Internals Program** procedures to take physical measurements of the Type 304 stainless steel hold-down springs in Unit 1 at each refueling outage to ensure preload is adequate for continued operation. B. Revise Reactor Vessel Internals Program procedures to include preload acceptance criteria for the Type 304 stainless steel hold-down springs in Unit 1.	SQN1: Prior to 09/17/20 SQN2: Not Applicable	B.1.34
28	A. Revise **Reactor Vessel Surveillance Program** procedures to consider the area outside the beltline such as nozzles, penetrations and discontinuities to determine if more restrictive pressure-temperature limits are required than would be determined by just considering the reactor vessel beltline materials. B. Revise Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM-E185-82 requirements, including the possibility of operation beyond 60 years (refer to the TVA Letter to NRC, "Sequoyah Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule Revision Due to License Renewal Amendment," dated January 10, 2013, ML13032A251.) C. Revise Reactor Vessel Surveillance Program procedures to withdraw and test a standby capsule to cover the peak fluence expected at the end of the period of extended operation.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.35

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
29	Implement the **Selective Leaching Program** as described in LRA Section B.1.37.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.37
30	Revise **Steam Generator Integrity Program** procedures to ensure that corrosion resistant materials are used for replacement steam generator tube plugs.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.39
31	A. Revise **Structures Monitoring Program** procedures to include the following in-scope structures: • Carbon dioxide building • Condensate storage tanks' (CSTs) foundations and pipe trench • East steam valve room Units 1 & 2 • Essential raw cooling water (ERCW) pumping station • High pressure fire protection (HPFP) pump house and water storage tanks' foundations • Radiation monitoring station (or particulate iodine and noble gas station) Units 1 & 2 • Service building • Skimmer wall (Cell No. 12) • Transformer and switchyard support structures and foundations B. Revise Structures Monitoring Program procedures to specify the following list of in-scope structures are included in the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program (Section B.1.36): • Condenser cooling water (CCW) pumping station (also known as intake pumping station) and retaining walls • CCW pumping station intake channel • ERCW discharge box • ERCW protective dike • ERCW pumping station and access cells • Skimmer wall, skimmer wall Dike A and underwater dam C. Revise Structures Monitoring Program procedures to include the following in-scope structural components and commodities: • Anchor bolts • Anchorage/embedments (e.g., plates, channels, unistrut, angles, other structural shapes) • Beams, columns and base plates (steel) • Beams, columns, floor slabs and interior walls (concrete) • Beams, columns, floor slabs and interior walls (reactor cavity and primary shield walls; pressurizer and reactor coolant pump compartments; refueling canal, steam generator compartments; crane wall and missile shield slabs and barriers) • Building concrete at locations of expansion and grouted anchors; grout pads for support base plates • Cable tray • Cable tunnel • Canal gate bulkhead • Compressible joints and seals	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.40

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(31)	• Concrete cover for the rock walls of approach channel • Concrete shield blocks • Conduit • Control rod drive missile shield • Control room ceiling support system • Curbs • Discharge box and foundation • Doors (including air locks and bulkhead doors) • Duct banks • Earthen embankment • Equipment pads/foundations • Explosion bolts (E. G. Smith aluminum bolts) • Exterior above and below grade; foundation (concrete) • Exterior concrete slabs (missile barrier) and concrete caps • Exterior walls: above and below grade (concrete) • Foundations: building, electrical components, switchyard, transformers, circuit breakers, tanks, etc. • Ice baskets • Ice baskets lattice support frames • Ice condenser support floor (concrete) • Insulation (fiberglass, calcium silicate) • Intermediate deck and top deck of ice condenser • Kick plates and curbs (steel - inside steel containment vessel) • Lower inlet doors (inside steel containment vessel) • Lower support structure structural steel: beams, columns, plates (inside steel containment vessel) • Manholes and handholes • Manways, hatches, manhole covers, and hatch covers (concrete) • Manways, hatches, manhole covers, and hatch covers (steel) • Masonry walls • Metal siding • Miscellaneous steel (decking, grating, handrails, ladders, platforms, enclosure plates, stairs, vents and louvers, framing steel, etc.) • Missile barriers/shields (concrete) • Missile barriers/shields (steel) • Monorails • Penetration seals • Penetration seals (steel end caps) • Penetration sleeves (mechanical and electrical not penetrating primary containment boundary) • Personnel access doors, equipment access floor hatch and escape hatches • Piles • Pipe tunnel • Precast bulkheads • Pressure relief or blowout panels • Racks, panels, cabinets and enclosures for electrical		

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(31)	equipment and instrumentation • Riprap • Rock embankment • Roof or floor decking • Roof membranes • Roof slabs • RWST rainwater diversion skirt • RWST storage basin • Seals and gaskets (doors, manways and hatches) • Seismic/expansion joint • Shield building concrete foundation, wall, tension ring beam and dome: interior, exterior above and below grade • Steel liner plate • Steel sheet piles • Structural bolting • Sumps (concrete) • Sumps (steel) • Sump liners (steel) • Sump screens • Support members; welds; bolted connections; support anchorages to building structure (e.g., non-ASME piping and components supports, conduit supports, cable tray supports, HVAC duct supports, instrument tubing supports, tube track supports, pipe whip restraints, jet impingement shields, masonry walls, racks, panels, cabinets and enclosures for electrical equipment and instrumentation) • Support pedestals (concrete) • Transmission, angle and pull-off towers • Trash racks • Trash racks associated structural support framing • Traveling screen casing and associated structural support framing • Trenches (concrete) • Tube track • Turning vanes • Vibration isolators D. Revise Structures Monitoring Program procedures to include periodic sampling and chemical analysis of ground water chemistry for pH, chlorides, and sulfates on a frequency of at least every five years. E. Revise Masonry Wall Program procedures to specify masonry walls located in the following in-scope structures are in the scope of the Masonry Wall Program: • Auxiliary building • Reactor building Units 1 & 2 • Control bay • ERCW pumping station • HPFP pump house • Turbine building		

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(31)	F. Revise Structures Monitoring Program procedures to include the following parameters to be monitored or inspected: • Requirements for concrete structures based on ACI 349-3R and ASCE 11 and include monitoring the surface condition for loss of material, loss of bond, increase in porosity and permeability, loss of strength, and reduction in concrete anchor capacity due to local concrete degradation. • Loose or missing nuts for structural bolting. • Monitoring gaps between the structural steel supports and masonry walls that could potentially affect wall qualification. G. Revise Structures Monitoring Program procedures to include the following components to be monitored for the associated parameters: • Anchors/fasteners (nuts and bolts) will be monitored for loose or missing nuts and/or bolts, and cracking of concrete around the anchor bolts. • Elastomeric vibration isolators and structural sealants will be monitored for cracking, loss of material, loss of sealing, and change in material properties (e.g., hardening). • Monitor the surface condition of insulation (fiberglass, calcium silicate) to identify exposure to moisture that can cause loss of insulation effectiveness. H. Revise Structures Monitoring Program procedures to include the following for detection of aging effects: • Inspection of structural bolting for loose or missing nuts. • Inspection of anchor bolts for loose or missing nuts and/or bolts, and cracking of concrete around the anchor bolts. • Inspection of elastomeric material for cracking, loss of material, loss of sealing, and change in material properties (e.g., hardening), and supplement inspection by feel or touch to detect hardening if the intended function of the elastomeric material is suspect. Include instructions to augment the visual examination of elastomeric material with physical manipulation of at least ten percent of available surface area. • Opportunistic inspections when normally inaccessible areas (e.g., high radiation areas, below grade concrete walls or foundations, buried or submerged structures) become accessible due to required plant activities. Additionally, inspections will be performed of inaccessible areas in environments where observed conditions in accessible areas exposed to the same environment indicate that significant degradation is occurring. • Inspection of submerged structures at least once every five years. Inspections of water control structures should be conducted under the direction of qualified personnel experienced in the investigation, design, construction, and operation of these types of facilities. • Inspections of water control structures shall be performed on an interval not to exceed five years.		

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(31)	• Perform special inspections of water control structures immediately (within 30 days) following the occurrence of significant natural phenomena, such as large floods, earthquakes, hurricanes, tornadoes, and intense local rainfalls. • Insulation (fiberglass, calcium silicate) will be monitored for loss of material and change in material properties due to potential exposure to moisture that can cause loss of insulation effectiveness. I. Revise Structures Monitoring Program procedures to prescribe quantitative acceptance criteria ~~is~~ based on the quantitative acceptance criteria of ACI 349.3R and information provided in industry codes, standards, and guidelines including ACI 318, ANSI/ASCE 11 and relevant AISC specifications. Industry and plant-specific operating experience will also be considered in the development of the acceptance criteria. J. Revise Structures Monitoring Program procedures to clarify that detection of aging effects will include the following. Qualifications of personnel conducting the inspections or testing and evaluation of structures and structural components meet the guidance in Chapter 7 of ACI 349.3R. K. Revise Structures Monitoring Program procedures to include the following acceptance criteria for insulation (calcium silicate and fiberglass) • No moisture or surface irregularities that indicate exposure to moisture. L. Revise Structures Monitoring Program procedures to include the following preventive actions. Specify protected storage requirements for high-strength fastener components (specifically ASTM A325 and A490 bolting). Storage of these fastener components shall include: 1) maintaining fastener components in closed containers to protect from dirt and corrosion; (2) storage of the closed containers in a protected shelter; (3) removal of fastener components from protected storage only as necessary; and (4) prompt return of any unused fastener components to protected storage.		
32	Implement the **Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)** as described in LRA Section B.1.41	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.41
33	A. Revise **Water Chemistry Control - Closed Treated Water Systems Program** procedures to provide a corrosion inhibitor for the following chilled water subsystems in accordance with industry guidelines and vendor recommendations: • Auxiliary building cooling • Incore Chiller 1A, 1B, 2A, & 2B • 6.9 kV Shutdown Board Room A & B	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.42

<table>
<tr><th>No.</th><th>COMMITMENT</th><th>IMPLEMENTATION SCHEDULE</th><th>LRA SECTION / AUDIT ITEM</th></tr>
<tr><td>(33)</td><td>B. Revise Water Chemistry Control - Closed Treated Water Systems Program procedures to conduct inspections whenever a boundary is opened for the following systems:
• Standby diesel generator jacket water subsystem
• Component cooling system
• Glycol cooling loop system
• High pressure fire protection diesel jacket water system
• Chilled water portion of miscellaneous HVAC systems (i.e., auxiliary building, Incore Chiller 1A, 1B, 2A, & 2B, and 6.9 kV Shutdown Board Room A & B)

C. Revise Water Chemistry Control-Closed Treated Water Systems Program procedures to state these inspections will be conducted in accordance with applicable ASME Code requirements, industry standards, or other plant-specific inspection and personnel qualification procedures that are capable of detecting corrosion or cracking.

D. Revise Water Chemistry Control - Closed Treated Water Systems Program procedures to perform sampling and analysis of the glycol cooling system per industry standards and in no case greater than quarterly unless justified with an additional analysis.

E. Revise Water Chemistry Control - Closed Treated Water Systems Program procedures to inspect a representative sample of piping and components at a frequency of once every ten years for the following systems:
• Standby diesel generator jacket water subsystem
• Component cooling system
• Glycol cooling loop system
• High pressure fire protection diesel jacket water system
• Chilled water portion of miscellaneous HVAC systems (i.e., auxiliary building, Incore Chiller 1A, 1B, 2A, & 2B, and 6.9 kV Shutdown Board Room A & B)

F. Components inspected will be those with the highest likelihood of corrosion or cracking. A representative sample is 20% of the population (defined as components having the same material, environment, and aging effect combination) with a maximum of 25 components. These inspections will be in accordance with applicable ASME Code requirements, industry standards, or other plant-specific inspection and personnel qualification procedures that ensure the capability of detecting corrosion or cracking.</td><td></td><td></td></tr>
<tr><td>34</td><td>Revise Containment Leak Rate Program procedures to require venting the SCV bottom liner plate weld leak test channels to the containment atmosphere prior to the CILRT and resealing the vent path after the CILRT to prevent moisture intrusion during plant operation.</td><td>SQN1: Prior to 09/17/20
SQN2: Prior to 09/15/21</td><td>B.1.7</td></tr>
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No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
35	Modify the configuration of the SQN Unit 1 test connection access boxes to prevent moisture intrusion to the leak test channels. Prior to installing this modification, TVA will perform remote visual examinations inside the leak test channels by inserting a borescope video probe through the test connection tubing.	SQN1: Prior to 09/17/20 SQN2: Not Applicable	B.1.6
36	Revise **Inservice Inspection Program** procedures to include a supplemental inspection of Class 1 CASS piping components that do not meet the materials selection criteria of NUREG-0313, Revision 2 with regard to ferrite and carbon content. An inspection techniques qualified by ASME or EPRI will be used to monitor cracking. Inspections will be conducted on a sampling basis. The extent of sampling will be based on the established method of inspection and industry operating experience and practices when the program is implemented, and will include components determined to be limiting from the standpoint of applied stress, operating time and environmental considerations.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.16
37	TVA will implement the **Operating Experience** for the **AMPs** in accordance with the TVA response to the RAI B.0.4-1 on July 29, 2013 letter to the NRC. (See Set 7.30day RAI B.0.4-1 Response, EDMS # L44130725002)	No later than the scheduled issue date of the renewed operating licenses for SQN Units 1 & 2.	B.0.4

The above table identifies the **37** SQN NRC LR commitments. Any other statements in this letter are provided for information purposes and are not considered to be regulatory commitments.