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

CNL-13-106

November 15, 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 13 (B.1.34-5a), 15 (4.2-1a), 16 (B.1.23-2c), 17 (B.1.23-2d), LRA Sections A.1.34 and B.1.34 Revision (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 13," dated September 16, 2013 (ADAMS Accession No. ML13256A007)
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 15," dated September 27, 2013 (ADAMS Accession No. ML13268A492)
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 16," dated October 18, 2013 (ADAMS Accession No. ML13282A330)
5. NRC Letter to TVA, "Requests for Additional Information for the Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application - Set 17," dated October 25, 2013 (ADAMS Accession No. ML13294A394)

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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.

By Reference 2, the NRC forwarded a request for additional information (RAI) labeled Set 13 that included RAI B.1.34-5a with a required response due date no later than November 15, 2013. By Reference 3, the NRC forwarded an RAI labeled Set 15 that included RAI 4.2-1a with a required response due date no later than November 26, 2013. By Reference 4, the NRC forwarded an RAI labeled Set 16 that included RAI B.1.23-2c with a required response due date no later than November 18, 2013. By Reference 5, the NRC forwarded an RAI labeled Set 17 that included RAI B.1.23-2d with a required response due date no later than November 25, 2013. Enclosure 1 provides the TVA response to these NRC RAIs. Enclosure 1 also includes revisions to LRA Sections A.1.34 and B.1.34.

Enclosure 2 provides an updated list of the regulatory commitments for license renewal that supersedes all previous versions.

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 15th day of November 2013.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:

1. TVA Responses to NRC Request for Additional Information: Sets 13 (B.1.34-5a), 15 (4.2-1a), 16 (B.1.23-2c), 17 (B.1.23-2d), LRA Sections A.1.34 and B.1.34 Revision
2. Regulatory Commitment List, Revision 12

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 13 (B.1.34-5a), 15 (4.2-1a), 16 (B.1.23-2c), 17 (B.1.23-2d), LRA Sections A.1.34 and B.1.34 Revision

<u>Set 13</u>: RAI B.1.34-5a

<u>Background:</u>

By letter dated August 9, 2013, the applicant provided its response to RAI B.1.34-5 that indicated the upper guide tube enclosure tubes, upper guide tube housing plate and upper instrumentation brackets, clamps, terminal [b]lock and conduit straps are potentially fabricated from cast austenitic stainless steel. The applicant provided the results of the failure modes, effects and criticality analysis (FMECA) conducted on the upper guide tube enclosure tubes, upper guide tube housing plate and upper instrumentation brackets, clamps, terminal [b]lock and conduit straps.

<u>Issue:</u>

The applicant indicated that after it considered the impact of the possible material changes to CASS it was concluded that these components remained in the "no additional measures" category and that the aging management strategy is not affected. However, the staff noted that the bases for applicant's conclusions from the FMECA of these CASS components were not provided in its response.

For each of these components (i.e., upper guide tube enclosure tubes, upper guide tube housing plate and upper instrumentation brackets, clamps, terminal [b]lock and conduit straps) the applicant indicated the likelihood of failure, likelihood of damage and FMECA Group based on the components being fabricated from an ASTM A351 Grade CF8 material; however, the technical basis that supports the new categorizations was not provided in the response to RAI B.1.34-5.

<u>Request:</u>

1. *Provide the technical basis for the FMECA conclusion that the CASS (1) upper guide tube enclosure tubes, (2) upper guide tube housing plate and (3) upper instrumentation brackets, clamps, terminal [b]lock and conduit straps components remained in the "no additional measures" inspection category.*
2. *Explain and justify the impact of considering loss of fracture toughness due to thermal embrittlement in the FMECA of the CASS (1) upper guide tube enclosure tubes, (2) upper guide tube housing plate and (3) upper instrumentation brackets, clamps, terminal [b]lock and conduit straps as compared to the original FMECA performed for MRP-227-A. In addition, specifically address how the stress and expected loading on these components was considered in the FMECA of these CASS components on the likelihood of damage and failure from cracking of potentially thermally embrittled components.*

TVA Response to RAI B.1.34-5a

1. The technical basis for the failure modes, effects, and criticality analysis (FMECA) conclusion that the cast austenitic stainless steel (CASS) (1) upper guide tubes, (2) upper guide tube housing plates and (3) upper instrumentation brackets, clamps, terminal blocks and conduit straps components remained in the "no additional measures" inspection category is that all components assigned to FMECA Groups 0 or 1 and for which there is low likelihood of failure are assigned to **Category A** per MRP-191 (Reference 1). The original FMECA, identified in MRP-227-A (Reference 2), for the subject components when fabricated from wrought 304 stainless steel (SS) follows the screening, categorization and ranking process described in MRP-191. Category A is defined as corresponding to those components that are below the screening criteria because aging degradation significance is minimal. Only the required ASME B&PV Code Section XI VT-3 examinations are necessary for Category A components. The potential for thermal embrittlement (TE) has a slight effect on the FMECA for these components, but the results of the revised FMECA, considering TE per MRP 191, still results in their categorization as Category A components, with recommendations for their inspection and evaluation unchanged from those of MRP-227-A.

   An expert panel was convened by Westinghouse to assess the effects of the use of ASTM A351 Grade CF8 CASS in the upper guide tube enclosures, upper guide tube housing plate and upper instrumentation conduit supports (including brackets, clamps, terminal blocks and conduit straps). **Table 1** provides the differences in the FMECA group for the subject components attributable to the use of CASS (as determined by the expert panel) instead of the use of wrought SS (as per MRP-191). The potential for TE as a degradation mechanism due to the use of CASS in these components results in the upper guide tubes, housing plates and the brackets, clamps, terminal blocks and conduit straps being placed in a higher FMECA group (i.e., group 1 rather than group 0). The upper guide tube enclosure tubes are retained in FMECA group 1 when the same change in material is taken into account.

   TABLE 1: FMECA Analyses for the Additional Components of Sequoyah (SQN) Units 1 and 2 Reactor Vessel Internals that may have been Fabricated from CASS ASTM A351 CF8

	Wrought ASTM A240 Type 304			**CASS ASTM A351 Grade CF8**		
Component	**Likelihood of Failure**	**Likelihood of Damage**	**FMECA Group**	**Likelihood of Failure**	**Likelihood of Damage**	**FMECA Group**
Upper guide tube enclosures	Low	Medium	1	Low	Medium	1
Housing plates	None		0	Low	Low	1
Brackets, terminal blocks and conduit straps	None		0	Low	Low	1

   The categorization process of MRP-191 assigns components in FMECA Group 0 and components in FMECA Group 1 with a low likelihood of failure to Category A for which "the aging degradation significance is minimal." MRP-227-A uses the results of the FMECA process to assign components to the functional groups "Primary," "Expansion," "Existing Programs" and "No Additional Measures." In the process used by MRP-227-A, components in Category A corresponding to those in FMECA Groups 0 and 1 per MRP-191 were placed in the functional group "No Additional Measures." Because the FMECA ranking of the upper guide tube enclosure tubes remains unchanged and the upper guide tubes were originally placed in the "No Additional Measures" functional group, this component set remains there. Because the only effect of potential TE on the categorization of the upper guide tube housing plate and the brackets, clamps, terminal blocks and conduit straps is to move them from FMECA Group 0 to Group 1 with a low likelihood of failure, they retain their screening categorization of Category A per MRP-191. Therefore, these components are retained in the functional group "No Additional Measures" per MRP-227-A.

2. As stated above, the FMECA groupings given in Table 1 were determined by an expert panel to assess the effects of the use of ASTM A351 Grade CF8 CASS in the upper guide tube enclosures, upper guide tube housing plates and upper instrumentation conduit supports (including brackets, clamps, terminal blocks and conduit straps). In MRP-191, these components were considered to be fabricated from wrought SS and, therefore, not susceptible to TE. Because of distance from the core, they were also not subject to sufficient neutron fluence to induce irradiation embrittlement. The expert panel considered how the parameters of the FMECA, the FMECA group, and the categorization for these components would be affected by the use of CASS and the potential for TE in these components. Because the consequences of damage effectively remain the same for these components, the key differences in the FMECA would be due to any potential increases in the likelihood of failure as a result of the use of the CASS material.

   The original determination of the FMECA, per MRP-191, resulted in these components being placed in FMECA Groups 0 and 1. The definitions of the categories of Likelihood of Component Failure and Likelihood of Consequences of Damage are those given in MRP-191. Categorization of the components for the Inspection and Evaluation Guidelines resulted in all of these components being placed in Category A. Category A corresponds to those components that are below the screening criteria so that aging degradation significance is minimal and for which only the required ASME B&PV Code Section XI VT-3 examinations are necessary. Category A essentially corresponds to those components placed in FMECA Groups 0 and 1, where those in FMECA Group 1 are also identified as having a low likelihood of failure. **Table 2** shows the original FMECA, per MRP-191, for the subject components.

Table 2: Original (MRP-191) Failure Modes Effects and Criticality Analysis for Questioned Upper Internals Assembly Components (originally designated as 304SS)

Assembly	Sub-assembly	Component	Material	IMT consequence of failure	Screened in degradation mechanism	Likelihood of failure L, M, H	Likelihood of damage L, M, H	FEMCA group
Upper Internals Assembly	Control rod guide tube assemblies and flow downcomers	Upper guide tube enclosure	304 SS	None[1]	SCC in weld	L (See a, in 'x' below)	M (See x)	1
		Housing plate	304 SS	G	None			0
	Upper instrument conduit and supports	Brackets, clamps, terminal blocks and conduit straps	304 SS	None[2]	None			0

The potential consequences of failures of these components were identified in MRP-191 as:

1 The consequence of failure of the upper guide tube enclosure would be misalignment of that guide tube with an impact on shutdown capability with respect to that particular control rod. Per the Issue Management Table, this failure has no significant consequence on the safe operation of the plant.

G The consequence of failure of the upper guide tube housing plates would be a loss of support for removable covers. Repair of this structure would incur significant economic impact, but there would be no effect on safe operation of the plant. Because there is no screened-in degradation mechanism for this component type, the potential for damage can be discounted.

2 The brackets, clamps, terminal blocks and conduit straps provide support to and position the core instrumentation (thermocouples). Loss of support would result in potential loss of individual thermocouple function and potential loose parts. In the absence of a screened in degradation mechanism, this likelihood of damage consequence is considered to be none for the 304 SS. Moreover, while the loss of support may induce malfunctioning of a thermocouple, because of the redundancy of thermocouples, disparities between sets of thermocouples would allow identification of such malfunctions. The consequence of this form of damage is, therefore, also considered insignificant.

x The consequence of failure of the upper guide tube enclosures is the potential misalignment of guide tubes impacting individual control rod function. This consequence of damage was evaluated as "medium" in MRP-191 because while the potential exist for single or multiple failures of control function, it was determined that there is sufficient redundancy to allow safe shutdown of the reactor.

MRP-191 also assessed the likelihood of failure of the upper guide tube enclosure as low (annotation "**a**") because there are no known failures in the existing fleet, and the most likely failure mode, via SCC of the welds, would result in relatively low stresses. Thus, the likelihood of failure within the period of extended operation (PEO) was considered low.

Based on the foregoing assessment, MRP-191 placed the components in FMECA Groups 0 and 1 where, in the latter case, the likelihood of failure is estimated to be low. According to the categorization process of MRP-191 (which was originally developed in MRP-134) components in these FMECA Groups are assigned as Category A components for evaluation and inspection. Per MRP-227-A, components in this category go into the "No Additional Measures" inspection category.

The expert panel assessed the effect on the FMECA and categorization of the components if they were fabricated from ASTM A351 Grade CF8 CASS. The expert panel determined that for the CASS material, TE became a potential aging degradation mechanism for these components. Irradiation embrittlement was also considered. The fluence experienced by these components, due to their location away from the high flux regions of the reactor, is below the threshold identified for IE of CASS in EPRI Report 1012081, *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175),* and is also below the 1×10^{17} n/cm^2 threshold identified in the NRC guideline for considering potential thermal irradiation synergy (Reference 3).

Based on this addition of TE as a potential aging degradation mechanism, the expert panel conducted a revised FMECA and Categorization taking into account that TE could be a potentially screened-in degradation mechanism, if these components were fabricated from ASTM 351 Grade CF8 CASS. The expert panel considered the potential for this mechanism to be screened in and the effects on the likelihood of failure and the likelihood of damage that would affect the FMECA and the resulting categorization of these components. This consideration produced the revised FMECA given in **Table 3**.

Table 3: Revised Failure Modes Effects and Criticality Analysis for Questioned Upper Internals Assembly Components Considering Potential Usage of ASTM A351 Grade CF8 CASS

Assembly	Sub-assembly	Component	Material	IMT consequence of failure	Screened in degradation mechanism	Likelihood of failure L, M, H	Likelihood of damage L, M, H	FEMCA group
Upper Internals Assembly	Control rod guide tube assemblies and flow downcomers	Upper guide tube enclosures	CF8	None[1]	SCC in weld, TE	L (See a)	M (See x)	1
		Housing plates	CF8	G	TE	L (See b)	L (See y)	1
	Upper instrument conduit and supports	Brackets, clamps, terminal blocks and conduit straps	CF8	None[2]	TE	L (See c)	L (See z)	1

The rationale for the determinations of the likelihood of failure and the likelihood of damage were identified as:

1 The major consequence of failure of the upper guide tube enclosures, whether fabricated from 304 SS or CASS, would be misalignment of that guide tube with potential impact on shutdown capability with respect to that particular control rod. The expert panel identified that the most likely mechanism of failure of the guide tube enclosures was still associated with failure of the tube enclosure flange welds that were included in consideration of the component when it was assumed to be fabricated from wrought SS. The expert panel concluded that failure due to TE of the CASS enclosures could occur, but that this was less likely than the previously identified failure mode of stress corrosion cracking of the weld. The effect of TE failure mode was considered in the assessment of the likelihood of damage.

a The likelihood of failure of the upper guide tube enclosures would not increase with the use of CASS and the associated potential for loss of toughness due to TE. The expert panel considered that the most likely mechanism of failure was still in the welded regions because the welds are the only portions of this component that are subject to stresses of sufficient level to result in failure. The expert panel considered that even with a potential loss of toughness due to TE of the CASS, the expert loading of this component provided insufficient stresses to result in fracture. Thus, the likelihood of failure of the upper guide tube enclosures remains "low" as defined in MRP-191.

x The consequence of failure of the upper guide tube enclosures is the potential misalignment of guide tubes impacting individual control rod function. This consequence of damage was evaluated as "medium" in MRP-191 for the considered failure of the upper guide tube flange welds. In these circumstances, while the potential exists for single or multiple failures of control function, there is sufficient redundancy to allow safe shutdown of the reactor. Use of CASS also introduced the potential for loss of material segments and production of loose parts if fracture could be result in sufficient tensile stresses in the thermally embrittled CASS sections. The production of loose parts was, however, identified by the expert panel as being of low consequence of damage. The consequence of damage for the additional degradation mechanism is lower than that assessed in MRP-191 for SCC of the welds in the component. That degradation mechanism was considered to have a "medium" likelihood of damage. Under these circumstances, the expert panel concluded that the likelihood of damage for the CASS upper guide tube enclosure was unchanged from that assessed in MRP-191 for the 304 SS component.

G The most significant consequence of failure of the upper guide tube housing plates, whether fabricated from CASS or Type 304 SS, would be a loss of support for removable covers. Repair of this structure would incur significant economic impact, but there would be no effect on the safe operation of the plant as a result of failure during operation. Per MRP-191, this consequence of damage still corresponds to a "low" likelihood of damage to the plant.

b The likelihood of failure of the housing plates was considered by the expert panel to be low. The expert panel considered that even in the reduced toughness condition potentially induced by TE, cracking was not a likely mode of failure because there were no significant driving stresses in this plate component. Given the absence of structural loading and the low flow conditions in the region of this component, the stresses imposed on the plate were considered insufficient to result in cracking.

y The likelihood of damage from potential failure due to cracking of the housing plates was considered by the expert panel to be low. The damage resulting from plate cracking was identified as the possibility for loose parts as well as the loss of local support for removable covers (which was previously identified in MRP-191 for a 304 SS housing plate). These loose parts were considered to have no significant effect on the ability to safely shutdown the reactor. Thus, the likelihood of plant damage due to failure of the CASS plate, taking into account the additional potential degradation mechanism of TE, was considered by the expert panel to be low.

2 The brackets, clamps, terminal blocks and conduit straps provide support to and position the core instrumentation (thermocouples). Loss of support would result in potential loss of individual thermocouple function and potential loose parts. The loss of support may induce malfunctioning of individual thermocouples. However, because of the redundancy of thermocouples, disparities between sets of thermocouples would allow identification of such malfunctions. Loose parts were also considered to have no significant effect on the ability to safely shutdown the reactor. Therefore, the consequence of loss of support or generation of loose parts is considered low.

c The likelihood of failure of the brackets, clamps, terminal blocks and conduits straps deriving from fracture due to loss of toughness resulting from TE is considered low because these components are operated under minimal loading. In the absence of significant tensile stresses, even in the potential presence of TE, fracture cannot occur. Therefore, the likelihood of failure of these components was considered by the expert panel to be low.

z The consequence of fracture of the brackets, clamps, terminal blocks and conduits straps would be potential loose parts and potential loss of thermocouple function. The loose parts would not have a significant effect on the ability to safely shutdown the reactor. The loss of support may also induce malfunctioning of individual thermocouples. However, because of the redundancy of thermocouples, disparities between sets of thermocouples would allow identification of such malfunctions. The expert panel considered that in comparison with MRP-191, the likelihood of damage was for the case of the use of CASS still considered low.

The foregoing discussion has specifically identified how the expected stress and loading of the components was considered in assessing the likelihood of damage and the likelihood of failure of the components from cracking due to TE. The most significant effect on the likelihood of damage was considered by the expert panel to be the additional potential for loose parts. This derives from the potential for embrittlement to occur in widespread locations of the CASS components. If cracking could be produced by sufficient tensile loading, then the resulting damage would be expected to include loose parts. The expert panel assessed the likelihood of plant damage as a result of these loose parts with respect to the MRP-191 criteria. The conclusion of this assessment was that the likelihood of damage was "low."

The expert panel also considered how the effect of the expected stress and loading on the components would affect the likelihood of failure of the components. The brackets, clamps, terminal blocks, and conduit straps and the housing plate all have low mechanical loads imposed on them and are not expected to have significant flow-induced loading. Thus, the stresses imposed on the material were considered by the expert panel to be insufficient to induce cracking even in thermally embrittled CASS. In the case of the upper guide tube enclosures, the expert panel considered that externally imposed stresses and loads in the flanges and tubes would be low and that the highest stresses would be those resulting from welding fabrication. These weld stresses were already identified in the FMECA of MRP-191 as driving the likelihood of failure of the upper guide tube enclosure when fabricated from wrought Type 304 SS. The expert panel concluded that these effects did not affect the FMECA groupings for the components and that the upper guide tube enclosures should be retained in FMECA Group 1 while the brackets, clamps, terminal blocks and conduit straps and the housing plate should be moved to FMECA Group 1 from FMECA Group 0.

<u>Response References</u>:

1. EPRI Report TR1013234, *Materials Reliability Program: Screening Categorization and Ranking of Reactor Vessel Internals Components for Westinghouse and Combustion Engineering PWR Designs (MRP-191)*
2. EPRI Report TR 1022863, *Materials Reliability Program: Pressurizer Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*
3. Refer to NRC License Renewal Issue N 98-0030, "Thermal Aging Embrittlement of CASS Components," C. I. Grimes, May 19, 2000. ADAMS No. ML003717179

<u>Set 15</u>: RAI 4.2-1a

<u>Background:</u>

In the applicant's response to RAI 4.2-1, Parts 1 – 4, dated August 9, 2013, the applicant states that the basis for the time-limited aging analysis (TLAAs) on reactor vessel (RV) neutron embrittlement are provided in Westinghouse Technical Report No. WCAP-17539.

In Footnote (f) of Table 3-1 of WCAP-17539, applicant identifies that it did not have any specific unirradiated upper shelf energy (UUSE) data for the fabrication of Unit 1 RV extended beltline circumferential weld W06 from Weld Heat No. 25006. The applicant therefore states that the UUSE value (i.e., 78 ft-lb) for Unit 1 RV weld W06 is based on Charpy impact tests from RV surveillance weld coupons (Surveillance Weld Heat No. 25295) that were removed and tested from Capsule T of the applicant's RV surveillance program.

In Footnote (f) of Table 3-2 of WCAP-17539, the applicant also identifies that it did not have specific UUSE data for the fabrication of RV circumferential welds W06 and W04 for Unit 2, which were fabricated from Weld Heat No. 721858. The applicant therefore states that the UUSE value for these welds (i.e., 78 ft-lb) is based on the limiting UUSE values for all RV beltline and extended beltline welds at Sequoyah Units 1 and 2. As a result, the UUSE value for Unit 2 RV extended beltline circumferential welds W04 and W06 is set to the UUSE value reported for Unit 1 RV extended beltline circumferential weld W06 (Weld Heat No. 25006).

<u>Issue:</u>

The UUSE values reported in the LRA for these RV extended beltline welds do not appear to be based on any NRC-endorsed positions on the establishment of UUSE values, such as those positions in NRC Generic Letter No. 92-01, Revision 1, or in NRC Branch Technical Position 5-3 NUREG 0800, "Standard Review Plan." As a result, the staff needs further demonstration that the UUSE values for Unit 1 RV extended beltline circumferential weld W06 and for Unit 2 RV extended beltline circumferential welds W06 and W04 are sufficiently conservative to ensure that either the upper shelf energy (USE) values for the welds will be maintained above a required end-of-life USE value of 50 ft-lb, in accordance with the USE requirements in 10 CFR Part 50, Appendix G, or that an appropriate equivalent margins analysis (EMA) has been performed for the welds.

<u>Request:</u>

1. *Provide your basis for why the UUSE value basis that has been reported for Unit 1 RV extended beltline circumferential weld W06 (i.e., 78 ft-lb), as reported in LRA Table 4.2-3 and based on Footnote (f) of Table 3-1 of WCAP-17539-NP, is considered to be a conservative basis for estimating the UUSE value of the component. In addition, justify how NRC Branch Technical Position 5-3 can be used as the basis for establishing a UUSE value of 78 ft-lbs for this weld when the weld heat identifier (Weld Heat No. 25006) is not represented in any of the surveillance capsules for the RV Surveillance Programs of Sequoyah Unit 1 or Unit 2.*
2. *Identify the NRC regulatory guidance or position that was used to establish a UUSE value of 78 ft-lb for Unit 2 RV extended beltline circumferential welds W06 and W04. Justify why the basis for establishing a UUSE value of 78 ft-lb for Unit 2 RV extended beltline circumferential welds W06 and W04, as reported in LRA Table 4.2-3 and based on Footnote (f) of Table 3-2 of WCAP-17539-NP, is considered sufficiently conservative and, therefore, a valid basis for the UUSE value, particularly in consideration of the fact that the weld heat identifier (i.e., Weld Heat No. 721858) is not represented in any of the RV Surveillance Programs for the Sequoyah units.*

3. *Given the absence of applicable UUSE data for Unit 1 RV extended beltline weld W06 and for Unit 2 RV extended beltline welds W06 and W04, justify why an EMA would not need to be performed for these welds as part of the LRA in order to demonstrate equivalency with the safety margin requirements in the ASME Code Section XI, Appendix G, as required by 10 CFR Part 50, Appendix G.*

TVA Response to RAI 4.2-1a

Request 1 and 2 Response:

10 CFR 50, Appendix G (Reference 1) requires that the upper-shelf energy (USE) of all reactor vessel beltline materials be measured before plant operation and that the licensee demonstrates that each material meets specific minimum values before and after exposure to fast neutron irradiation. The USE toughness requirements of Appendix G, based on Charpy impact tests, is 75 ft-lbs minimum before and 50 ft-lb minimum after irradiation. The requirement for irradiated USE can be met based on post-irradiation surveillance program measurements of the surveillance materials or based on predictions, but the latter requires the availability of an initial USE. The practice of developing a full Charpy transition curve for all beltline welds was not implemented until after Appendix G was first issued in 1973. Prior to that time, the ASME Code requirement was to perform three Charpy tests at -12°C/10°F and measure only absorbed energy for a given set of weld consumables. Therefore, insufficient data exists for determining an initial USE value for many welds which were deposited prior to the implementation of the Appendix G requirements. In numerous instances, full Charpy curves were developed for test plate welds and reactor pressure vessel surveillance welds. Full Charpy curves are available for weld materials qualified subsequent to the issuance of Appendix G requirements. Generic initial USE values and/or estimation methodologies would need to be developed in order to demonstrate compliance for those cases in which the data were never generated.

The general practice instituted by various vessel manufacturers in response to Appendix G was to perform weld material certification tests for each set of weld consumables, i.e., produce a weldment with a specific heat(s) of weld wire and lot of flux, perform a simulated post-weld heat treatment on it, machine test specimens from it, test the Charpy specimens to develop a full Charpy curve, and perform the other required tests. This practice was also followed prior to 1973 for many weld test plates and surveillance program weldments. Therefore, the older data could serve as the certification tests for other welds made with the same weld wire heat and lot of flux. This approach was used in numerous cases to comply with the Appendix G requirements for initial USE. In certain cases, the USE data were historically applied to welds fabricated with the same wire heat and using an equivalent welding process but a different lot of flux. This was considered to be reasonable because an "equivalent" weld process would entail the use of the same types of consumables, meaning the same type of wire and flux. A logical extension of this was to use the available data to establish generic values for a group of welds fabricated using an equivalent process. Because no generic USE values are available for the type of weld material used in fabrication of the SQN reactor vessel welds, the best available data were used to establish reasonable, yet conservative, initial USE values for weld heat numbers 25006 and 721858.

The weld qualification record, dated 1969, for weld heat # 25006 (SQN, Unit 1 Upper Shell (US) to Intermediate Shell (IS) Circumferential (Circ.) Weld W06) reports two sets of three Charpy impact energy values at a single test temperature (-12°C /10°F) with no reported shear data. No other Charpy impact energy information is available for this weld heat. As described above, this level of Charpy testing is consistent for non-beltline welds fabricated before 1973, such as this weld for the SQN, Unit 1 reactor vessel. The average Charpy impact energy value of the six available data points was 59.7 ft-lbs and the highest reported impact energy value tested at -12°C /10°F was 71.2 ft-lbs (See Table 1).

The weld qualification record, dated 1969, for weld heat # 721858 (SQN, Unit 2 US to IS Circ. Weld W06 and Lower Shell (LS) to Bottom Head Ring (BHR) Circ. Weld W04) reports only three Charpy impact energy values at a single test temperature (-12°C /10°F) with no reported shear data. No other Charpy impact energy information is available for this weld heat. Consistent with SQN, Unit 1, the average Charpy impact energy value of the three available data points was 68.3 ft-lbs and the highest reported impact energy value tested at -12°C /10°F was also 71.2 ft-lbs (See Table 2).

Table 1: Summary of Available Charpy Impact Energy Data for SQN, Unit 1 Weld Heat No. 25006

Material & ID #	Heat #	Temperature (°F/°C)	Average Charpy Energy (ft-lb)	Max Charpy Energy (ft-lb)
US to IS Circ Weld W06	25006	10 / -12	59.7	71.2

Table 2: Summary of Available Charpy Impact Energy Data for SQN, Unit 2 Weld Heat No. 721858

Material & ID #	Heat #	Temperature (°F/°C)	Average Charpy Energy (ft-lb)	Max Charpy Energy (ft-lb)
US to IS Circ Weld W06 and LS to BHR Weld W04	721858	10 / -12	68.3	71.2

Because testing only occurred at 10°F for weld heat numbers 25006 and 721858, the actual initial USE value for these weld heats will be higher than the maximum Charpy impact energy value of 71.2 ft-lbs achieved during Charpy testing of each weld. Therefore, in order to justify a higher, more metallurgically realistic initial USE value for SQN, Units 1 and 2, Paragraph 2, Section B.1.2 of NUREG-0800 Branch Technical Position 5-3 (Reference 2) was applied to the SQN, Units 1 and 2 reactor vessel welds. Reference 2 states:

> *"If Charpy upper shelf energy values were not obtained, conservative estimates should be made using results of tests on specimens from the first surveillance capsule removed."*

With consideration that Reference 2 makes no mention of material heat numbers and in the absence of USE data for weld heat # 25006, the weld heat # 25295 test results from the first surveillance capsule were used in accordance with Section B.1.2 of Reference 2. The SQN, Unit 1 surveillance weld heat # 25295 is a Rotterdam weld of the same type (SMIT 40 with SMIT 89 flux) that was welded at approximately the same time at the same shop as weld heat # 25006. The irradiated USE value from Capsule T, i.e., the first surveillance capsule pulled and tested from SQN, Unit 1, is 78 ft-lbs and was taken from Table C-1 of WCAP-15224, Revision 0 (Reference 3). TVA considers this value to be a reasonable and conservative estimate of the initial USE for heat # 25006, because the highest reported Charpy impact energy value tested at -12°C /10°F was 71.2 ft-lbs, (see Table 1).

Similarly to SQN, Unit 1, in the absence of USE data for weld heat # 721858, the weld heat # 4278 test results from the first surveillance capsule were used for SQN, Unit 2, in accordance with Section B.1.2 of Reference 2. The SQN, Unit 2 surveillance weld heat # 4278 is also a Rotterdam weld of the same type (Arcos with SMIT 89 flux type) that was welded at approximately the same time at the same shop as weld heat # 721858. The irradiated USE value from Capsule T, the first surveillance capsule pulled and tested from SQN, Unit 2, is 110 ft-lbs and was taken from Table C-1 of WCAP-15320, Revision 0 (Reference 4). Because this irradiated USE value is actually higher than the unirradiated value for the reactor vessel weld metal (102 ft-lbs for heat # 4278 – see Table 4), the lowest initial USE value from the SQN, Units 1 and 2 welds was conservatively assumed to be the initial USE value for heat # 721858. This initial USE is 78 ft-lbs, which is associated with the SQN, Unit 1 US to IS circ. weld W06 and the irradiated USE value taken from the surveillance weld material from Capsule T. TVA considers this value a reasonable and conservative estimate of the initial USE for heat # 721858, because the highest reported Charpy impact energy value for weld heat # 721858 tested at -12°C /10°F was 71.2 ft-lbs (see Table 2).

The full Charpy plots of the SQN, Units 1 and 2 surveillance weld materials, heat numbers 25295 and 4278, were reviewed to determine whether the average USE values at 10°F for weld heat numbers 25006 and 721858 are consistent with those seen for the surveillance welds. Test results documented in WCAP-15224 for SQN, Unit 1 surveillance weld heat # 25295 and WCAP-15320 for SQN, Unit 2 surveillance weld heat # 4278 at 20°F and 0°F are summarized in Table 3. Note that these temperatures were chosen for comparison purposes because 10°F test results were not available for the surveillance weld materials. The average energy values of 76.7 and 63.7 ft-lbs for heat numbers 25295 and 4278 are consistent with the average energy values of 59.8 and 69.3 ft-lbs for the two reactor vessel extended beltline weld materials. It was also observed from the review of the unirradiated surveillance welds that the Charpy upper shelf energies began to form at temperatures greater than 100°F; further validating that at 10°F, the Charpy impact energy levels for weld heat numbers 25006 and 721858 were not near their ultimate initial USE value.

Table 3: Charpy Impact Energy Data for SQN, Units 1 and 2 Surveillance Welds at Temperatures Comparable to Tests Performed on Weld Heat Numbers 25006 and 721858

Plant	Surveillance Weld Heat #	Temperature (°F)	Average Charpy Energy (ft-lb)
SQN, Unit 1	25295	20	76.7
SQN, Unit 2	4278	0	63.7

To substantiate that the establishment of the Charpy USE value from the first capsule withdrawn from SQN, Units 1 and 2 as the initial USE value for weld heat numbers 25006 and 721858 is a conservative estimate, the available reactor vessel beltline and surveillance weld initial USE values, and first irradiated capsule USE values, from the domestic Westinghouse-design plants that have a reactor vessel fabricated by Rotterdam were reviewed. A mean minus 2-sigma value, calculated from the initial USE values from all of the reactor vessel beltline and surveillance weld materials, could be used to establish a lower bound USE consistent with the approach used for the Combustion Engineering Linde welds (Flux types 1092, 0091 and 124) in CEN-622-A (Reference 5.) Therefore, a comparison was made between the mean minus 2-sigma USE value and the 78 ft-lbs initial USE determined for the SQN welds. Additionally, a review of the surveillance weld USE values from the first withdrawn irradiated capsule was performed to assure that a lower bound initial USE value was used for SQN, Unit 1 and 2 weld heat numbers 25006 and 721858, respectively. Domestic Westinghouse-design reactor vessel (RV) and surveillance weld data from SQN, Units 1 and 2, North Anna, Units 1 and 2, McGuire, Unit 2, Catawba, Unit 1, and Watts Bar, Unit 1 were considered in this assessment. Rotterdam used an assortment of weld vendors which used flux types SMIT 89, GRAU LO and Linde LW320 for these reactor vessel welds. The results of this review are documented in Table 4.

Table 4: Summary of all Available Charpy Initial USE Values for Rotterdam Reactor Vessel Beltline and Surveillance Welds

Plant	Heat #	Weld Type	Initial Unirradiated USE (ft-lb)	First Irradiated Capsule USE (ft-lb)	Data Source
SQN, Unit 1	25295	RV	113	---	Table 3-1 in Ref. 6
		Surveillance	111	78	Table C-1 in Ref. 3
SQN, Unit 2	4278	RV	102	---	Table 3-2 in Ref. 6
		Surveillance	112	110	Table C-1 in Ref. 4
North Anna Unit 1	25531	RV	102	---	Ref. 7
		Surveillance	95	92	Ref. 7
North Anna Unit 2	716126	RV	107	---	Ref. 7
		Surveillance	112	92	Ref. 7
McGuire Unit 2	895075	RV	132	---	Table 4.2-2 in Ref. 8
		Surveillance	133	138	App. C in Ref. 9
Catawba Unit 1	895075	RV	130	---	Table 4.2-3 in Ref. 8
		Surveillance	129	123	Table C-1 in Ref. 10
Watts Bar Unit 1	895075	RV	134	---	Ref. 7
		Surveillance	131	143	Table C-1 in Ref. 11

Based on the reactor vessel beltline and surveillance weld data presented in Table 4, the average initial USE value across all seven operating U.S. Westinghouse-design plant reactor vessels fabricated by Rotterdam is 117.4 ft-lbs, with a standard deviation of 13.6 ft-lbs. Therefore, a mean minus 2-sigma lower bound initial USE value is 90.1 ft-lbs. The initial USE value previously established for the two SQN welds, heat numbers 25006 and 721858, was 78 ft-lbs, using the NUREG-0800 approximation methodology based on results from the first capsule tested at SQN, Unit 1. This is a conservative approach compared to the mean minus 2-sigma method, which was used as a basis for the generic USE values developed in CEN-622-A for the Combustion Engineering Linde welds (Flux type 1092, 0091 and 124). Also note that the value of 78 ft-lbs is the lowest irradiated first capsule USE value across all seven reactor vessels that have been put into service.

In summary, due to a lack of a full Charpy impact energy curve, an initial USE value was estimated for SQN, Units 1 and 2 weld heat numbers 25006 and 721858, respectively, based on the guidance provided in Reference 2. TVA considers the initial USE value of 78 ft-lbs for these weld heats to be a reasonable and conservative estimate of the true initial USE value had either of these welds been tested to achieve a full Charpy curve.

Request 3 Response:

Based on the above, an equivalent margins analysis (EMA) is not required for weld heat numbers 25006 and 721858 at SQN, Units 1 and 2 respectively, because the initial USE value has been conservatively established as 78 ft-lbs, which is greater than the 10 CFR 50 Appendix G requirement of 75 ft-lbs. In addition, the requirements for the minimum initial USE value, which was developed circa 1973, postdate the welding and qualification testing of these weldments at Rotterdam. The weld qualification records for weld heat numbers 25006 and 721858 at SQN, Units 1 and 2 are both dated 1969 as mentioned above and were not considered for a full Charpy test because they were both outside the traditional beltline region. As documented in WCAP-17539-NP (Reference 6), both welds remain above the 10 CFR 50, Appendix G limit of 50 ft-lbs, after taking into consideration IE through the end of the extended operating term of 52 effective full-power years (EFPY).

Therefore, an EMA is not necessary for SQN, Units 1 and 2 because the margins of safety to the 10 CFR 50, Appendix G USE limits are maintained through the end of extended operation.

Response References:

1. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.
2. Branch Technical Position 5-3, Revision 2, "Fracture Toughness Requirements," Contained in Chapter 5 of Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, NUREG-0800, March 2007.
3. Westinghouse Report, WCAP-15224, Revision 0, "Analysis of Capsule Y from the Tennessee Valley Authority SQN, Unit 1 Reactor Vessel Radiation Surveillance Program," June 1999.
4. Westinghouse Report, WCAP-15320, Revision 0, "Analysis of Capsule Y from the Tennessee Valley Authority SQN, Unit 2 Reactor Vessel Radiation Surveillance Program," December 1999.

5. Combustion Engineering Report CEN-622-A, "Generic Upper Shelf Values for Linde 1092, 124 and 0091 Reactor Vessel Welds," CEOG Task 839, C-E Owners Group, December 1996.

6. Westinghouse Report, WCAP-17539-NP, Revision 0, "SQN, Units 1 and 2 Time-Limited Aging Analysis on Reactor Vessel Integrity," March 2012.

7. Nuclear Regulatory Commission Reactor Vessel Integrity Database (RVID), Version 2.0.1, July 6 2000.

8. McGuire Units 1 and 2 and Catawba Units 1 and 2 License Renewal Application, "Application to Renew Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2," Duke Energy, June 2001.

9. Westinghouse Report, WCAP-14799, "Analysis of Capsule W from the Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program," March 1997.

10. Westinghouse Report, WCAP-15117, "Analysis of Capsule V and the Dosimeters from Capsules U and X from the Catawba Unit 1 Reactor Vessel Radiation Surveillance Program," October 1998.

11. Westinghouse Report, WCAP-16760, "Analysis of Capsule Z from the Tennessee Valley Authority, Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program," November 2007.

Set 16: RAI B.1.23-2c

Background:

In its September 30, 2013, response to RAI B.1.23-2a, the applicant described justification for why an inspection program is not necessary to manage the wear of the control rod drive mechanism (CRDM) nozzles. As part of the response, the applicant addressed its analysis on the maximum wear depth of CRDM nozzles (i.e., 0.050 inches).

In its response, the applicant stated that, when contact occurs between the CRDM nozzle and the thermal sleeve centering pads of the nozzle, only a relatively small wear volume of the three centering pads is distributed over the relatively large areas of the CRDM nozzle inside surfaces. The applicant acknowledged that the specific hardness values of the sleeves, centering pads and CRDM nozzles are not known, and further stated that similar grades of stainless steel and Inconel materials have similar hardness values (i.e., approximately 90 on Rockwell 8 scale).

Issue:

As previously discussed in RAI B.1.23-2a, the applicant's analysis on the maximum wear depth involves uncertainties in local vibratory motions, residual stresses, and hardness levels of the CRDM nozzles, thermal sleeves, and centering pads. Without an inspection of the wear indications, localized severe wear conditions cannot be excluded. Therefore, the staff finds that an inspection program is necessary to confirm the adequacy of the applicant's analysis.

In addition, wear of the CRDM nozzles may interfere with the volumetric examination of the CRDM nozzles which is specified in the applicant's Nickel Alloy Inspection Program. The staff needs to clarify how the applicant's program would resolve the situation that wear of the CRDM nozzles interferes with the volumetric examination of the CRDM nozzles.

Request:

1. *Identify an inspection program to confirm the adequacy of the applicant's analysis on the maximum wear depth of the CRDM nozzles. As part of the response, describe how applicant's inspection program confirms the adequacy of the applicant's analysis.*
2. *Clarify whether the applicant's Nickel Alloy Inspection Program accounts for a potential loss of ultrasonic testing signal due to the surface irregularities of the wear areas.*
3. *In addition, describe how the applicant's Nickel Alloy Inspection Program would resolve the interference caused by wear of the CRDM nozzles with volumetric examination of the CRDM nozzles. As part of the response, clarify how the applicant's program will confirm the absence of cracking in the wear areas that could not be adequately examined by the volumetric examination.*

TVA Response to RAI B.1.23-2c

1. As a part of the existing ASME Section XI Code Case N-729 augmented inspection at SQN, the inside diameters of CRDM housing penetrations (also referred to as CRDM adapters and CRDM nozzles) with thermal sleeve centering pads in the weld examination volume will be inspected for evidence of thinning at the centering pad locations. This inspection will be performed as part of the Inservice Inspection Program described in LRA Appendix B, Section B.1.16. and Commitment **#36.D**
   During the 2007 ASME Code Case N-729 inspections, the wear groove indications on the center most CRDM housing penetrations at SQN were observed, but not measured, using angle beam ultrasonic testing (UT) scans of the examination volume. Operability determination was based on wear depth analysis.

TVA continues to evaluate industry operating experience related to CRDM housing penetration wear and initiatives to measure CRDM housing penetration wear and resulting wall thickness. Upon successful demonstration of a wear depth measurement process, SQN will use the demonstrated process at accessible locations to measure depth of wear on the CRDM housing penetration wall associated with contact with the CRDM thermal sleeve centering pads. Commitment **#36.C** has been added.

The depth of wear at the accessible CRDM housing penetration locations is expected to represent the greatest wear from the thermal sleeve centering pads at all of the wear locations. The centermost CRDM housing penetrations are at approximately the top dead center of the head. Therefore, they have the longest thermal sleeve length exposed to fluid flow forces and have higher centering pad contact forces. At other CRDM housing penetration locations, a shorter thermal sleeve length is exposed to the fluid flow force; thus, reduced centering pad wear. Since the CRDM thermal sleeves are supported from the top of the CRDM adapters, the centermost pad wear at the thermal sleeve lower pads will bound that at the upper pads.

The depth of wear determined from this inspection would be used to confirm the adequacy of the design basis analyses and be used to estimate the projected wear at the end of the next inspection interval.

2. The inspection of the CRDM housing penetrations is a qualified process that accounts for surface irregularities associated with wear from the CRDM thermal sleeve centering pads. For clarity, this inspection is now reflected as part of the Inservice Inspection Program described in LRA B.1.16.

3. The qualified UT examination procedures and equipment used on the nickel alloy SQN CRDM housing penetrations have been demonstrated capable of detecting cracking on an intact cylindrical bore that has wear caused from contacting with the CRDM thermal sleeve centering pads.

   Based on industry field examination data, the centering pad wear is not uniform around the circumference, but eccentric. To evaluate the effectiveness of the procedure in adapting to the centering pad wear geometry, two mockups were fabricated to replicate the centering pad wear conditions (i.e., a cylindrical CRDM housing penetration inside diameter (ID) with an off-center circle (diameter slightly larger than the ID) worn into the ID surface of the CRDM housing penetration). One mockup was fabricated from Alloy 600, and the second mockup was fabricated from Alloy 690. Both mockups have a carbon steel ring shrunk fit onto the penetration outer diameter (OD) to simulate the upper head shell.

   The Alloy 600 in the first mockup was fabricated with a centering pad wear depth range from 0.010 inches to 0.065 inches with a symmetrical wear cross section. The mockup contains 28 ID and OD, axial and circumferential electrical discharge machined (EDM) notches placed within and adjacent to the centering pad wear locations. The mockup also contains three uniform machined channels in the "shrink fit" region of the carbon steel ring material to simulate leak paths. The intent of the mockup was to determine if the technique is limited by a specific wear depth and to evaluate the effects of the centering pad wear grooves have on the ultrasonic leak path signature.

   The Alloy 690 in the second mockup was fabricated with centering pad wear depth of 0.025 inches with an asymmetrical wear cross section. The mockup contains 18 ID and OD, axial and circumferential and off-axis hot isostatic pressed EDM notches placed on and adjacent to centering pad wear locations. In addition, the mockup contains two leak paths that traverse the centering pad wear grooves. The intent of the second mockup is the same as the intent of the first mockup.

The cylinder to the centering pad wear transition causes probe "lift-off" as the probe moves from the cylindrical surface to the wear surface. The probe "lift-off" results in a shift (water delay) of the data at the cylindrical wear section transition, wear section, and transition from wear section to cylindrical section of the housing penetration. However, the ability of the coolant to "flood" the region provides adequate coupling, which enhances UT signal quality. Therefore, the results of UT using the mockups described above confirmed the ability of the procedure to compensate for the interference caused by wear of the CRDM housing penetrations during the UT examination of the CRDM housing penetrations and ensure effective flaw detection and sizing.

The SQN-qualified UT examination procedures and equipment have demonstrated the capability for detecting cracking in the CRDM housing penetration wear area in the examination volume. Accordingly, the susceptible examination volume required by ASME Code Case N-729-1 is examined for cracking using this qualified technique, which accounts for potential transducer lift-off.

Changes to **LRA Section A.1.16,** Inservice Inspection Program, follow with additions underlined.

"The Inservice Inspection Program will be enhanced as follows.

<u>Revise Inservice Inspection Program procedure to monitor the wear of the accessible CRDM housing penetrations in weld examination volume.</u>

<u>Revise Inservice Inspection Program procedure to perform an examination of the accessible CRDM housing penetration to determine the amount of wear in the area of the thermal sleeve centering pads for Units 1 and 2. The accessible locations consist of the centermost CRDM housing penetrations 1 through 5.</u>

<u>Revise Inservice Inspection Program procedure to estimate the wall thickness at the end of the next RVH inspection interval and compare this projected wall thickness to the thickness used in Sequoyah design basis analyses to demonstrate validity of the analyses</u>."

(Commitments **#36.F.D.E.**)

Changes to **LRA Section B.1.16** follow with additions underlined.

"Enhancements

The following enhancements will be implemented prior to the PEO.

Element Affected	Enhancement
3. Parameters Monitored or Inspected	Revise Inservice Inspection Program procedure to monitor the wear of the accessible CRDM housing penetrations in weld examination volume. (Commitment **#36.F**)
4. .Detection of Aging Effects	Revise Inservice Inspection Program procedure to perform an examination of the accessible CRDM housing penetrations to determine the amount of wear in the area of the thermal sleeve centering pads for Units 1 and 2. The accessible locations consist of the centermost CRDM housing penetrations 1 through 5. (Commitment **#36.D**)
5. Monitoring and Trending	Revise Inservice Inspection Program procedure to estimate the CRDM housing penetration wear at the end of the next RVH inspection interval and compare the projected wall thickness to the thickness used in Sequoyah design basis analyses to demonstrate validity of the analyses. (Commitment **#36.E**)

Changes to **LRA Table 3.1.2-1** line items follow with additions underlined.

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG -1801 Item	Table 1 Item	Notes
Closure head • CRDM housing penetration	Pressure boundary	Nickel alloy	Treated borated water (int)	Loss of material due to wear	Inservice Inspection Program			H

"

Commitments **#36.C.D.E.F** have been added

Set 17: RAI B.1.23-2d

Background:

By letter dated October 17, 2013, the applicant responded to request for additional information (RAI) B.1.23-2b, which addressed the applicant's operating experience regarding loss of material due to wear in thermal sleeves of control rod drive mechanism (CRDM) nozzles. In its response, the applicant revised its license renewal application (LRA) to identify loss of material and cracking as aging effects requiring management for the thermal sleeves. The applicant's response also indicates the applicant uses its Inservice Inspection Program to manage loss of material and cracking for the CRDM nozzle thermal sleeves.

The applicant further stated that in parallel with the volumetric examination or surface examination performed on the CRDM nozzles, the thermal sleeves of CRDM nozzles are examined for loss of material in accordance with Westinghouse Technical Bulletin TB-07-2. In addition, the applicant stated that the thermal sleeve examination inspects the area where the thermal sleeves penetrate underneath the reactor vessel head in the two outer most concentric rows of CRDM nozzles. The applicant also stated that areas of loss of material are identified and documented in the Corrective Action Program.

Issue:

The staff noted that the applicant's response does not address the following items which are necessary to demonstrate the adequacy of applicant's aging management for loss of material and cracking of the CRDM nozzle thermal sleeves.

- *Inspection methods to detect loss of material and cracking*
- *Inspection frequencies to manage loss of material and cracking*
- *Total number of thermal sleeves in the two outer most concentric rows of CRDM nozzles for each unit (as baseline information)*
- *How the applicant's program confirms that loss of material and cracking are not occurring in thermal sleeves which are not located in the two outer most concentric rows of CRDM nozzles.*

Request:

As discussed in the issue section of this RAI, identify the following items: (1) inspection methods to detect loss of material and cracking; (2) inspection frequencies to manage loss of material and cracking; (3) total number of thermal sleeves in the two outer most concentric rows of CRDM nozzles for each unit; and (4) how the applicant's program confirms that loss of material and cracking are not occurring in thermal sleeves which are not located in the two outer most concentric rows of CRDM nozzles.

In addition, justify why the applicant's inspection methods, frequencies and scopes are adequate to manage loss of material and cracking for the CRDM nozzle thermal sleeves.

TVA Response to RAI B.1.23-2d

Item 1, Inspection methods:

During the 2007 Reactor Vessel Head (RVH) inspection 100 percent of the CRDM thermal sleeves were visually inspected where the thermal sleeve penetrates underneath the RVH for cracking and loss of material due to wear. During the upcoming outages for SQN, Units 1 and 2, the same locations will be visually inspected. In addition, the six locations exhibiting the greatest amount of wear will be examined with a 0 degree ultrasonic technique, or equivalent, to measure actual wall thickness of the thermal sleeve at the identified locations.

Item 2, Inspection frequencies:

The CRDM thermal sleeve inspections at SQN are augmented examinations in the SQN ISI Program. The CRDM thermal sleeve inspections are performed at the same frequency as the RVH volumetric exams, in accordance with ASME Code Case N-729-1.

Item 3, Total number of thermal sleeves in the two outermost concentric rows of CRDM nozzles for each unit:

There are sixteen thermal sleeves in the two outermost concentric rows of CRDM nozzles in each unit.

Item 4: Confirmation of the adequacy of thermal sleeves at nozzle locations other than the two outermost concentric rows:

The initial SQN inspection in 2007 examined all of the CRDM nozzles, not just those in the two outermost concentric rows. No cracking was identified, but the thermal sleeves at nozzles 20, 39, 40, 45, 47, 51, 53 and 57 were noted as having the most wear. The maximum wear was identified on the CRDM thermal sleeve at nozzle 20.

The engineering evaluation of the observed wear of the thermal sleeve at nozzle 20 consisted of a comparison to wear observed at another facility. The observed wear at SQN was less than the wear at the other facility. On this basis, the most significant measured wear at the other facility was used to evaluate remaining service life of thermal sleeves at SQN. Conservatively using this measured wear for SQN and the Reference 1 wear evaluation report, a minimum projected remaining life of 21.6 EPFY was determined for the most severely worn thermal sleeve. Because this projected life is much longer than the required inspection interval for the RVH inspection, the frequency as provided for in the augmented inspection under the SQN ISI Program is adequate to manage loss of material and cracking of the thermal sleeves.

To supplement the CRDM thermal sleeve inspections, SQN plans confirmatory thermal sleeve wall thickness inspections. The results of these inspections will confirm the adequacy of the existing SQN analysis.

Changes to **LRA Section A.1.16** follow with additions underlined.

"Program Description

The Inservice Inspection Program manages loss of material, cracking, thermal embrittlement, flaw growth, and reduction in fracture toughness for ASME Class 1, 2, and 3 pressure-retaining components, including welds, pump casings, valve bodies, integral attachments, and pressure retaining bolting using volumetric, surface, and/or visual examination and leakage testing of ASME Class 1, 2 and 3 component as specified in ASME Section XI code, 2001 Edition 2003 addendum. Additional limitations, modifications, and augmentations described in 10 CFR 50.55a are included as a part of this program. Every ten years this program is updated to the latest ASME Section XI code edition and addendum approved by the NRC in 10 CFR 50.55a. Repair and replacement activities for these components are covered in Subsection IWA of the ASME code edition of record.

The Inservice Inspection Program will be enhanced as follows.

Revise the Inservice Inspection Program procedures to perform an augmented visual inspection of the Unit 1 and Unit 2 CRDM thermal sleeves and a wall thickness measurement of the six thermal sleeves exhibiting the greatest amount of wear. The results of the augmented inspection should be used to project if there is sufficient wall thickness for the PEO, or until the next inspection." (Commitment **#36.B**)

Changes to **LRA Section B.1.16** follow with additions underlined.

"Program Description

The Inservice Inspection Program manages loss of material, cracking, thermal embrittlement, flaw growth, and reduction in fracture toughness for ASME Class 1, 2, and 3 pressure-retaining components, including welds, pump casings, valve bodies, integral attachments, and pressure retaining bolting using volumetric, surface, and/or visual examination and leakage testing of ASME Class 1, 2 and 3 component as specified in ASME Section XI code, 2001 Edition 2003 addendum. Additional limitations, modifications, and augmentations described in 10 CFR 50.55a are included as a part of this program. Every ten years this program is updated to the latest ASME Section XI code edition and addendum approved by the NRC in 10 CFR 50.55a. Repair and replacement activities for these components are covered in Subsection IWA of the ASME code edition of record.

Enhancements

The following enhancement will be implemented prior to the PEO.

Element Affected	Enhancement
4. Detection of Aging Effect	Revise the Inservice Inspection Program procedures to perform an augmented visual inspection of the Unit 1 and Unit 2 CRDM thermal sleeves and a wall thickness measurement of the six thermal sleeves exhibiting the greatest amount of wear. The results of the augmented inspection should be used to project if there is sufficient wall thickness for the PEO, or until the next inspection. (Commitment **#36.B**)

"

<u>Revision to LRA Sections A.1.34 and B.1.34</u>:

MRP-227-A, PWR Reactor Internals Inspection and Evaluation Guidelines, Safety Evaluation Report was approved by the NRC in June 22, 2012, (ADAMS No. ML111600498).

MRP-227-A (table 4-3, page 4-28, ADAMS No. ML12108A217) states:

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Expansion Method Frequency (Note 1)	Examination Coverage
Alignment and interfacing components Internals hold down spring	All Plants with 304 Stainless Steel hold down springs	Distortion (loss of load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms. [7]	None	Direct measurement of spring height within *three cycles of the beginning of the license renewal period*. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty.

SQN LRA Appendix C, Table C-1, states:

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	**Expansion Method Frequency** (Note 1)	Examination Coverage
Alignment and interfacing components • Internals hold down spring	SQN Unit 1	Distortion (loss of load) (Note 7)	None	Direct measurement of spring height within *three cycles of the beginning of the license renewal period*. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty.

In the SQN LRA, as submitted on January 7, 2013, (ADAMS No. ML13024A004), LRA Sections A.1.34 and B.1.34 enhancements are inconsistent with the SQN LRA Appendix C, Table C-1, "Expansion Method Frequency" column. TVA revises Commitment **#27.A, LRA Sections A.1.34** and **B.1.34**. with additions underlined and deletions lined through.

LRA Sections A.1.34

"A.1.34 Reactor Vessel Internals Program

The Reactor Vessel Internals Program will be enhanced as follows.

- Revise **Reactor Vessel Internals Program** procedures to <u>perform direct measurement of Unit 1 304 SS hold down spring height within three cycles of the beginning of the period of extended operation. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.</u>
~~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.~~

- Revise Reactor Vessel Internals Program procedures to include preload acceptance criteria for the Type 304 stainless steel hold-down spring in Unit 1.

Enhancement~~s~~ will be implemented prior to the period of extended operation."

LRA Section B.1.34

"<u>Enhancements</u>

The following enhancement will be implemented prior to the period of extended operation.

Element Affected	Enhancement
4. Detection of Aging Effects	Revise **Reactor Vessel Internals Program** procedures to <u>perform direct measurement of Unit 1 304 SS hold down spring height within three cycles of the beginning of the period of extended operation. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.</u> ~~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.~~
6. Acceptance Criteria	Revise Reactor Vessel Internals Program procedures to include preload acceptance criteria for the Type 304 stainless steel hold-down spring in Unit 1.

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Commitment **#27.A** is revised.

ENCLOSURE 2

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

Regulatory Commitment List, Revision 12

Commitments **27.A** and **36.B.C.D.E.F** have been revised.

This Commitment Revision supersedes all previous versions. The latest revision will be included in the **LRA Appendix A**, before the SQN LRA SER is issued.

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 (PEO) 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 PEO 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	6.A,B,C,E: 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 specify the following for insulated components. • Periodic representative inspections are conducted during each 10-year period beginning 5 years before the PEO. • For a representative sample of outdoor components, except tanks, and indoor components, except tanks, identified with more than nominal degradation on the exterior of the component, insulation is removed for visual inspection of the component surface. Inspections include a minimum of 20 percent of the in-scope piping length for each material type (e.g., steel, stainless steel, copper alloy, aluminum). For components with a configuration which does not conform to a 1-foot axial length determination (e.g., valve, accumulator), 20 percent of the surface area is inspected. Inspected components are 20% of the population of each material type with a maximum of 25. Alternatively, insulation is removed and component inspections performed for any combination of a minimum of 25 1-foot axial length sections and individual components for each material type (e.g., steel, stainless steel, copper alloy, aluminum.) • For a representative sample of indoor components, except tanks, operated below the dew point, which have not been identified with more than nominal degradation on the exterior of the component, the insulation exterior surface or jacketing is inspected. These visual inspections verify that the jacketing and insulation is in good condition. The number of representative *jacketing inspections will be at least 50 during each 10-year* period. If the inspection determines there are gaps in the insulation or damage to the jacketing that would allow moisture to get behind the insulation, then removal of the insulation is required to inspect the component surface for degradation. • For a representative sample of indoor insulated tanks operated below the dew point and all insulated outdoor tanks, insulation is removed from either 25 1-square foot sections or 20 percent of the surface area for inspections of the exterior surface of each tank. The sample inspection points are distributed so that inspections occur on the tank dome, sides, near the bottom, at points where structural supports or instrument nozzles penetrate the insulation, and where water collects (for example on top of stiffening rings).	6.D: SQN1: Prior to 09/17/15 SQN2: Prior to 09/15/16	

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM

(6)	• Inspection locations are based on the likelihood of corrosion under insulation (CUI). For example, CUI is more likely for components experiencing alternate wetting and drying in environments where trace contaminants could be present and for components that operate for long periods of time below the dew point.
	• If tightly adhering insulation is installed, this insulation should be impermeable to moisture and there should be no evidence of damage to the moisture barrier. Given that the likelihood of CUI is low for tightly adhering insulation, a minimal number of inspections of the external moisture barrier of this type of insulation, although not zero, will be credited toward the sample population.
	• Subsequent inspections will consist of an examination of the exterior surface of the insulation for indications of damage to the jacketing or protective outer layer of the insulation, if the following conditions are verified in the initial inspection.
	  • No loss of material due to general, pitting or crevice corrosion, beyond that which could have been present during initial construction
	  • No evidence of cracking
	Nominal degradation is defined as no loss of material due to general, pitting, or crevice corrosion, beyond that which could have been present during initial construction, and no evidence of cracking. If the external visual inspections of the insulation reveal damage to the exterior surface of the insulation or there is evidence of water intrusion through the insulation (e.g. water seepage through insulation seams/joints), periodic inspections under the insulation will continue as described above. [RAI 3.0.3-1 Request 6]
	
	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.
	
	F. For a representative sample of outdoor insulated components and indoor insulated components operated below the dew point, which have been identified with more than nominal degradation on the exterior of the component, insulation is removed for inspection of the		

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(6)	component surface. For a representative sample of indoor insulated components operated below the dew point, which have not been identified with more than nominal degradation on the exterior of the component, the insulation exterior surface is inspected. These inspections will be conducted during each 10-year period beginning **5** years before the PEO. [RAI 3.0.3-1 Request 6]	6.F: SQN1: Prior to 09/17/15 SQN2: Prior to 09/15/16	
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 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 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

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
9	Implement the **Fire Water System Program** as described in LRA Section B.1.13. 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 PEO 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 PEO. 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 PEO and periodically during the PEO based on the findings from the inspections performed prior to the PEO. C. Revise Fire Water System Program procedures to ensure ~~a~~ sprinkler heads are tested in accordance with NFPA-25 (2011 Edition), Section 5.3.1 [RAI 3.0.3-1 Request 4] 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. F. Prior to the PEO, SQN will select an inspection method (or methods) that will provide suitable indication of piping wall thickness for a representative sample of buried piping locations to supplement the existing inspection locations for high pressure fire protection system 26 and essential raw cooling water system 67. [RAI 3.0.3-1, request 5a, Set 10.30, 9/3/13] G. Revise Fire Water System Program procedures to-periodically remove a representative sample of components such as sprinkler	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.13

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(9)	heads or couplings prior to the PEO and perform a visual internal inspection of dry fire water system piping for evidence of corrosion, loss of wall thickness, and foreign material that may result in flow blockage using the methodology described in NFPA-25 Section 14.2.1. This includes those sections of dry piping described in NRC Information Notice (IN) 2013-06, where drainage is not occurring. The acceptance criteria shall be "no debris" (i.e., no corrosion products that could impede flow or cause downstream components to become clogged). Any additional inspections in accordance with NFPA-25, Sections 14.2.1 or 14.2.2 will be based on the initial inspection results. H. Revise Fire Water System Program procedures to perform an obstruction evaluation in accordance with NFPA-25 (2011 Edition), Section 14.3.1. I. Revise Fire Water System Program procedures to conduct follow-up volumetric examinations if internal visual inspections detect surface irregularities that could be indicative of wall loss below nominal pipe wall thickness. J. Revise Fire Water System Program procedures to annually inspect the fire water storage tank exterior painted surface for signs of degradation. If degradation is identified, conduct follow-up volumetric examinations to ensure wall thickness is equal to or exceeds nominal wall thickness. The fire water storage tanks will be inspected in accordance with NFPA-25 (2011 Edition) requirements. K. Revise Fire Water System Program procedures to include a fire water storage tank interior inspection every five years that includes inspections for signs of pitting, spalling, rot, waste material and debris, and aquatic growth. Include in the revision direction to perform fire water storage tank interior coating testing, if any degradation is identified, in accordance with ASTM D 3359 or equivalent, a dry film thickness test at random locations to determine overall coating thickness; and a wet sponge test to detect pinholes, cracks or other compromises of the coating. If there is evidence of pitting or corrosion ensure the Fire Water System Program procedures direct performance of an examination to determine wall and bottom thickness. L. Revise Fire Water System Program procedures based on the results of a feasibility study to perform the main drain tests in accordance with NFPA-25 (2011 Edition) Section 13.2.5. M. Revise Fire Water System Program procedures to perform spray head discharge pattern tests from all open spray nozzles to ensure that patterns are not impeded by plugged nozzles, to ensure that nozzles are correctly positioned, and to ensure that obstructions do not prevent discharge patterns from wetting surfaces to be protected.		

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(9)	Where the nature of the protected property is such that water cannot be discharged, the nozzles shall be inspected for proper orientation and the system tested with smoke or some other medium to ensure that the nozzles are not obstructed. [RAI 3.0.3-1, Request 4, for Commitments 9.C,G to M]		
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
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.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.18

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(13)	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		
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
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 PEO 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 A. TVA response to **RAI B.1.25.1a** 1. Repair the manhole sump pump and discharge piping deficiencies associated with the accumulation of water in seven manholes/handholes that are scheduled for correction and/or mitigation by September 2015. (HH3, HH2B, HH52B, HH55A2, MH7B, MH10A and MH32B as identified on October 1, 2013) 2. Grade the ground surface around Manhole 31 to direct runoff away from the manhole. The re-grading is scheduled for completion by September 2014. 3. Prior to the PEO, the license renewal commitment for the Non-EQ Inaccessible Power Cables (400 V to 35 kV) Program will establish diagnostic testing activities on all inaccessible power	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21 18.A.1: Sept 2015 18.A2 & 4: Sept 2014 18.A.3: SQN1: Prior to 09/17/20	B.1.25

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(18)	cables in the 400 V to 35kV range that are in the scope of license renewal and subject to aging management review. 4. Revise the manhole inspection procedures to specify the maximum allowable water level to preclude cable submergence in the manhole. If the inspection identifies submergence of inaccessible power cable for more than a few days, the condition will be documented and evaluated in the SQN corrective action program. The evaluation will consider results of the most recent diagnostic testing, insulation type, submergence level, voltage level, energization cycle (usage), and various other inputs to determine whether the cables remain capable of performing their intended current licensing basis function.	SQN2: Prior to 09/15/21	
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	A. 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. B. RAI 3.0.3-1, Request 3, Loss of Coating Integrity: For in-scope components that have internal Service Level III or Other coatings, initial inspections will begin no later than the last scheduled refueling outage prior to the PEO. Subsequent inspections will be performed based on the initial inspection results.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21 24.B SQN1: RFO Prior to 09/17/20 SQN2: RFO Prior to 09/15/21	B.1.31
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.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.32

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(25)	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.		
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 perform direct measurement of Unit 1 304 SS hold down spring height within three cycles of the beginning of the period of extended operation. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years. (11/15/13 Letter, Enclosure 1, pages 24-25)~~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~~ Within three U1 refuel cycles of the date 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 PEO.	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 • Concrete cover for the rock walls of approach channel	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 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 equipment and instrumentation		

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(31)	• 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. M. From **RAI B.1.40-4a** Response (Turbine Building wall crack) 1. SQN will map and trend the crack in the condenser pit north wall. 2. SQN will test water inleakage samples from the turbine building condenser pit walls and floor slab for minerals and iron content to assess the effect of the water inleakage on the concrete and the reinforcing steel. 3. SQN will test concrete core samples removed from the turbine building condenser pit north wall with a minimum of one core sample in the area of the crack. The core samples will be tested		

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(31)	for compressive strength and modulus of elasticity and subjected to petrographic examination. 4. The results of the tests and SMP inspections will be used to determine further corrective actions, if necessary. 5. Commitment #31.M will be implemented before the PEO for SQN Units 1 and 2.		
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 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)	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.42

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
(33)	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.		
34	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.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.7
35	A. From **RAI B.1.6-1** Response: 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. B. From B.1.6-1b Response: To monitor the condition of the access boxes and associated materials, perform visual examinations of all accessible surfaces, including the access box surfaces, cover plate, welds, and gasket sealing surfaces of the access boxes on each unit every other refueling outage with the gasketed access box lid removed. [RAI B.1.6-1b] C. From B.1.6-2b Response: Continue volumetric examinations where the SCV domes were cut at the frequency of once every five years until the coatings are reinstalled at these locations. [RAI B.1.6-2b]	35.A: SQN1: Prior to 09/17/20 SQN2: Not Applicable 35. B & C: SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.6

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
36	A. 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. B. Revise the Inservice Inspection Program procedures to perform an augmented visual inspection of the Unit 1 and Unit 2 CRDM thermal sleeves and a wall thickness measurement of the six thermal sleeves exhibiting the greatest amount of wear. The results of the augmented inspection should be used to project if there is sufficient wall thickness for the PEO, or until the next inspection. (RAI B.1.23-2d) C. Evaluate industry operating experience related to CRDM housing penetration wear and initiatives to measure CRDM housing penetration wear and resulting wall thickness. Upon successful demonstration of a wear depth measurement process, SQN will use the demonstrated process at accessible locations to measure depth of wear on the CRDM housing penetration wall associated with contact with the CRDM thermal sleeve centering pads. (RAI B.1.23-2c) D. Revise Inservice Inspection Program procedure to perform an examination of the accessible CRDM housing penetrations to determine the amount of wear in the area of the thermal sleeve centering pads for Units 1 and 2. The accessible locations consist of the centermost CRDM housing penetrations 1 through 5. (RAI B.1.23-2c) E. Revise Inservice Inspection Program procedure to estimate the CRDM housing penetration wear at the end of the next RVH inspection interval and compare the projected wall thickness to the thickness used in Sequoyah design basis analyses to demonstrate validity of the analyses. (RAI B.1.23-2c) F. Revise Inservice Inspection Program procedure to monitor the wear of the accessible CRDM housing penetrations in weld examination volume. (RAI B.1.23-2c)	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.16

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	LRA SECTION / AUDIT ITEM
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, ML13213A027); and Oct 16, 2013 2013 letter to the NRC. (See Set 13.30d RAIs B.0.4-1a and A.1-1a Response) • Revise OE Program Procedure to include current and future revisions to NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," as a source of industry OE, and unanticipated age-related degradation or impacts to aging management activities as a screening attribute. • Revise the CAP Procedure to provide a screening process of corrective action documents for aging management items, the assignment of aging corrective actions to appropriate AMP owners, and consideration of the aging management trend code. • Revise AMP procedures as needed to provide for review and evaluation by AMP owners of data from inspections, tests, analyses or AMP OEs. • Revise the OE Program Procedure to provide guidance for reporting plant-specific OE on unanticipated age-related degradation or impact to aging management activities to the TVA fleet and/or INPO. • Revise the OE, CAP, Initial and Continuing Engineering Support Personnel Training to address age-related topics, the unanticipated degradation or impacts to the aging management activities; including periodic refresher/update training and provisions to accommodate the turnover of plant personnel, and recent AMP-related OE from INPO, the NRC, Scientech, and nuclear industry-initiated guidance documents and standards." • A comprehensive and holistic AMP training topic list will be developed before the date the SQN renewed operating license is scheduled to be issued. • TVA AMP OE Process, AMP adverse trending & evaluation in CAP, AMP Initial and Refresher Training will be fully implemented by the date the SQN renewed operating license is scheduled to be issued.	No later than the scheduled issue date of the renewed operating licenses for SQN Units 1 & 2. (Currently February 2015)	B.0.4
38	Implement the **Service Water Program** as described in LRA Section B.1.38. (RAI 3.0.3-1, Request 3)	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21	B.1.38

The above table identifies the <u>38</u> SQN NRC LR commitments. Any other statements in this letter are provided for information purposes and are not considered to be regulatory commitments.

This Commitment Revision supersedes all previous versions.