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

June 25, 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 Reactor Vessel Internals Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application, Set 1 (TAC Nos. MF0481 and MF0482)**

References:
1. TVA 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," dated April 26, 2013 (ADAMS Accession No. ML13109A515)

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 license for the Sequoyah Nuclear Plant, Units 1 and 2. The request would extend the license for an additional 20 years beyond the current expiration date. By letter dated April 26, 2013 (Reference 2), the NRC forwarded a request for additional information (RAI). The required date for the response is within 60 days of the date stated in the RAI, i.e., no later than June 25, 2013.

Enclosure 1 to this letter provides TVA's response to the Reference 2 RAI, except for question B.1.34-5. Mr. Richard Plasse, the NRC License Renewal Project Manager, has given a verbal extension to August 9, 2013 for this question.

Enclosure 2 is an updated listing of the regulatory commitments for license renewal. The sole change is additional clarification to Commitment 7.B.

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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 25th day of June 2013.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:
1. TVA Responses to NRC Request for Additional Information
2. Regulatory Commitment List, Revision 2

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

ENCLOSURE 1

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

TVA Responses to NRC Request for Additional Information

RAI B.1.34-1
Background:
License renewal application (LRA) Section B.1.34 provides enhancements to the "detection of aging effects" and "acceptance criteria" program elements of the Reactor Vessel Internals Program. These enhancements are associated with revising the program procedures to account for taking physical measurements, including the preload acceptance criteria, for the Type 304 stainless steel hold-down spring in Unit 1.

Applicant/Licensee Action Item (A/LAI) No. 5 of MRP-227-A states, in part, that applicants/licensees ***shall*** *identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs. It also states, in part, that the applicant/licensee* ***shall*** *include its proposed acceptance criteria with an explanation of how the functionality of the component being inspected will be maintained under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227.*

The applicant's response A/LAI No. 5 in LRA Appendix C states that the plant specific acceptance criteria for hold-down springs and an explanation of how the proposed acceptance criteria are consistent with the Sequoyah Nuclear Plant (SQN) licensing basis and the need to maintain the functionality of the hold-down springs under all licensing basis conditions will be developed prior to the first required physical measurement.

Issue:
A/LAI No. 5 requires the identification of the plant-specific acceptance criteria to be applied when performing the physical measurements and an explanation of how the functionality of the component being inspected will be maintained under all licensing basis conditions of operation during the period of extended operation.

The applicant's proposed enhancements to revise its procedures to take physical measurements of the Type 304 stainless steel hold-down spring in Unit 1 and to include preload acceptance criteria does not adequately address A/LAI No. 5 of MRP-227-A. Specifically, the applicant did not provide its plant-specific acceptance criteria for the Type 304 stainless steel hold-down spring in Unit 1 and the explanation outlined in A/LAI No. 5.

Request:

- *Define and justify the physical measurement techniques that will be used to determine RVI hold-down spring height when inspections are performed on the component in accordance with the MRP-227-A.*

  Explain and justify how the proposed acceptance criteria is consistent with the Unit 1 licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation.

- *Revise the response to A/LAI No. 5, as necessary.*

TVA Response to RAI B.1.34-1

- In accordance with MRP-227-A, an inspection of the Sequoyah Nuclear Plant (SQN) Unit 1 reactor vessel internals (RVI) Type 304 stainless steel hold-down spring is required to ensure that there is no unacceptable loss of preload. The physical measurement technique used will be a direct measurement of the RVI hold-down spring height with the spring lying flat on the surface of the core barrel flange. The direct measurement will be made under water using long handle tools with calibrated measurement instrumentation. Three measurements will be taken every 45 degrees around the circumference of the spring to minimize uncertainty. The location of the measurement point at the top surface of the core barrel hold-down spring will be based on positioning the tool front face in contact with the outer diameter of the spring.

  The acceptance criterion is the measured height of the spring as a function of time relative to the required hold-down force. The decrease in hold-down spring height is assumed to occur linearly over time. The approach used to develop the hold-down spring height acceptance criterion is to consider the actual hold-down spring height at plant start-up and the hold-down spring height required at the end of 60 years to provide adequate hold-down force. A linear interpolation at the time of the hold-down spring height measurement determines the required minimum hold-down spring height.

  Applicable plant loading conditions consistent with the Unit 1 licensing basis were evaluated to determine the hold-down force necessary to maintain functionality. Details for the hold-down spring height measurements, acceptance criteria, and confirmatory actions, if applicable, are summarized in a Westinghouse proprietary calculation. Hold-down spring height measurements less than the required minimum hold-down spring height indicate a need for re-evaluation and successive measurement or a replacement hold-down spring.

- The change to the response to Applicant/Licensee Action Item (A/LAI) No. 5 in LRA Appendix C is with additions underlined and deletions shown with strikethrough.

  ~~"The SQN plant specific acceptance criteria for hold-down springs and an explanation of how the proposed acceptance criteria are consistent with the SQN licensing basis and the need to maintain the functionality of the hold-down springs under all licensing basis conditions will be developed prior to the first required physical measurement.~~

  In accordance with MRP-227-A, an inspection of the Unit 1 reactor vessel internals (RVI) Type 304 stainless steel hold-down spring is required to ensure that there is no unacceptable loss of preload. The physical measurement technique used will be a direct measurement of the reactor vessel internals hold-down spring height with the spring lying flat on the surface of core barrel flange. The direct measurement will be made under water using long handle tools with calibrated measurement instrumentation. Three measurements will be taken every 45 degrees around the circumference of the spring to minimize uncertainty. The location of the measurement point at the top surface of the core barrel hold-down spring will be based on positioning the tool front face in contact with the outer diameter of the spring.

  The acceptance criterion is the measured height of the spring as a function of time relative to the required hold-down force. The decrease in hold-down spring height is assumed to occur linearly over time. The approach used to develop the hold-down spring height acceptance criterion is to consider the actual hold-down spring height at plant start-up and the hold-down spring height required at the end of 60 years to provide adequate hold-down force. A linear interpolation at the time of the hold-down spring height measurement determines the required minimum hold-down spring height.

  Applicable plant loading conditions consistent with the Unit 1 licensing basis were evaluated to determine the hold-down force necessary to maintain functionality. Details for the hold-down spring height measurements, acceptance criteria, and confirmatory actions, if applicable, are summarized in a Westinghouse proprietary calculation. Hold-down spring height measurements less than the required minimum hold-down spring height indicate a need for re-evaluation and successive measurement or a replacement hold-down spring."

RAI B.1.34-2

<u>Background</u>

LRA Section 3.1.2.2.9.B.2 states that Unit 2 uses a Type 403 stainless steel hold-down spring. LRA Table 3.1.2-2 indicates that the "interfacing components: internals hold-down spring" made of stainless steel (Type 403) is a "No Additional Measures" component for "loss of material – wear" and "loss of preload" as part of the Reactor Vessel Internals Program.

A/LAI No. 5 of MRP-227-A states, in part, that applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs. It also states, in part, that the applicant/licensee shall include its proposed acceptance criteria and an explanation of how the functionality of the component being inspected will be maintained under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227.

<u>Issue</u>:

The Westinghouse Type 403 stainless steel hold down spring is not specifically excluded from the scope of A/LAI No. 5 for MRP-227-A. Since the Type 403 stainless steel hold down spring was not addressed in the MRP-227-A (staff reviewed and approved), it is not clear to the staff why the applicant did not address this component in its response to A/LAI No. 5 in LRA Appendix C or justify that this component does not need to managed for "loss of material – wear" and "loss of preload."

<u>Request</u>:

- *Justify that the Unit 2 Type 403 stainless steel hold down spring is not subject to stress relaxation such that the functionality of the component will be maintained under all licensing basis conditions of operation during the period of extended operation.*

- *In lieu of this demonstration, revise the Reactor Vessels Internals Program, LRA Table 3.1.2-2 and LRA Table C-1 to identify that the Unit 2 hold down springs made of Type 403 stainless steel are managed for "loss of material – wear" and "loss of preload" as a "Primary" component.*

  In addition, provide responses to the following questions, as discussed in A/LAI No. 5 of MRP-227-A:
  - *Define and justify the physical measurement techniques that will be used to determine RVI hold-down spring height when inspections are performed on the component in accordance with the MRP-227-A.*
  - *Explain and justify how the proposed acceptance criteria is consistent with the Unit 2 licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation.*

- *Revise the response to A/LAI No. 5, as necessary.*

TVA Response to RAI B.1.34-2

- The Westinghouse evaluation of the Unit 2 Type 403 stainless steel hold-down spring has concluded that the Type 403 stainless steel hold-down spring is not subject to stress relaxation such that the functionality of the component will be jeopardized under all licensing basis conditions of operation during the period of extended operation.

  The Materials Reliability Program (MRP) 191 specifically considered the stress relaxation of Type 403 stainless steel hold-down springs. The behavior of Type 403 stainless steel was considered significantly improved over that of Type 304 stainless steel, which is an alternative hold-down spring material, and has provided acceptable operating experience over many years. There are no documented reports on the measurements of stress relaxation behavior of Type 304 or Type 403 stainless steels at the temperatures of interest (i.e., around 300°C). However, as documented in the Westinghouse evaluation, the stress relaxation of Type 403 stainless steel at 400°C has been observed to be significantly lower than the stress relaxation of Type 304 stainless steels at the same temperature. Because stress relaxation is a thermally activated process, the stress relaxation differences between Type 304 and 403 stainless steels at 400°C will be maintained at the lower temperatures relevant to hold-down spring service.

  The Type 403 stainless steel hold-down spring is expected to continue to exhibit less stress relaxation than would be observed in the Type 304 stainless steel hold-down spring. Therefore, the Type 403 stainless steel hold-down spring is expected to provide adequate resistance to stress relaxation and maintain its functionality for Unit 2 licensing basis conditions over the period of extended operation. On this basis, the Type 403 hold-down springs were screened out for stress relaxation effects in MRP-191. The MRP concluded that inspections of the hold-down springs were not warranted for MRP-227-A.

- Based on the above response, there is no need to revise LRA Table 3.1.2-2 or the Reactor Vessel Internals Program described in LRA Section B.1.34. The change to the Applicability and Effect (Mechanism) columns of LRA Appendix C, Table C-1, page C-16, is with additions underlined and deletions shown with strikethrough.

Applicability	Effect (Mechanism)
SQN Unit 1 <u>Type 304 SS</u>	<u>For Type 304 SS</u> ~~D~~distortion (loss of load) (Note 7)
<u>SQN Unit 2 Type 403 SS</u>	<u>For Type 403 SS distortion (loss of load) is insignificant due to material properties at PWR operating temperatures (see response to A/LAI #5)</u>.

- Based on the above discussion, the change to the response to Applicant/Licensee Action Item (A/LAI) No. 5 is to include the following paragraphs in addition to the revision described in the response to RAI B.1.34-1.

  "The materials reliability project (MRP) specifically considered the stress relaxation of Type 403 stainless steel hold-down springs. The behavior of Type 403 stainless steel was considered significantly improved over that of Type 304 stainless steel, which is an alternative hold-down spring material and has provided acceptable operating experience over many years. There are no documented reports on the measurements of stress relaxation behavior of Type 304 or Type 403 stainless steels at the temperatures of interest (i.e., around 300°C). However, the stress relaxation of Type 403 stainless steel at 400°C has been observed to be significantly lower than the stress relaxation of Type 304 stainless steels at the same temperature. Because stress relaxation is a thermally activated process, the stress relaxation difference of Type 304 and 403 stainless steels at 400° C will be maintained at the lower temperatures relevant to hold-down spring service.

  The Type 403 stainless steel hold-down spring is expected to continue to exhibit less stress relaxation than would be observed in the Type 304 stainless steel hold-down spring. Therefore, the Type 403 stainless steel hold-down spring is expected to provide adequate resistance to stress relaxation and maintain its functionality for SQN Unit 2 licensing basis conditions over the period of extended operation. On this basis, the Type 403 hold-down springs were screened out for stress relaxation effects in MRP-191. The MRP concluded that inspections of the hold-down springs were not warranted for MRP-227-A."

RAI B.1.34-3

Background:

A/LAI No. 8 states, in part, for those cumulative usage factor (CUF) analyses that are TLAAs for reactor vessel internals, the acceptance of these TLAAs may be done in accordance with either 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with 10 CFR 54.21(c)(1)(iii) using the applicant's program that corresponds to NUREG-1801, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program." To satisfy the evaluation requirements of ASME Code, Section III, Subsection NG-2160 and NG-3121, the existing fatigue CUF analyses shall include the effects of the reactor coolant system water environment. The applicant's response to Part 5 of A/LAI No. 8 in LRA Appendix C states that TLAAs are identified in LRA Section 4.

LRA Section 4.3.1.2 provides the applicant's TLAA for reactor vessel internal components with CUF values, which include the lower core plate and control rod drive guide tube pins. The applicant dispositioned this TLAA in accordance with 10 CFR 54.21(c)(1)(iii) such that the Fatigue Monitoring Program will manage the effects of aging due to fatigue on the reactor vessel internals.

Issue:

Since the TLAA is managed with the Fatigue Monitoring Program, the staff noted that LRA Section 4.3 and LRA Appendix C do not address the aspect in Part 5 of A/LAI No. 8 that states "the existing fatigue CUF analyses shall include the effects of the reactor coolant system water environment."

Request:

> *Since the Fatigue Monitoring Program will manage the effects of aging due to fatigue on the reactor vessel internals in accordance with 10 CFR 54.21(c)(1)(iii), justify how the existing fatigue CUF analyses will include the effects of the reactor coolant system water environment as discussed in Part 5 of A/LAI No. 8. Revise the response to A/LAI No. 8 and LRA Sections A.1.11 and B.1.11 to explicitly describe how the effects of the reactor coolant system water environment for the reactor vessel internals TLAA will be managed.*

TVA Response to RAI B.1.34-3

The existing cumulative usage factor (CUF) analyses for the RVIs do not include the effects of the reactor coolant system (RCS) water environment. However, SQN will revise the CUF analyses for the RVIs (lower core plate and control rod drive (CRD) guide tube pins) to account for the effects of the RCS water environment prior to the period of extended operation.

The change to the response to Applicant/Licensee Action Item (A/LAI) #8 part 5 provided in LRA Appendix C is with additions underlined. The change to the second enhancement in the program description in LRA Appendix A, Section A.1.11, and LRA Appendix B, Section B.1.11, is shown below with additions underlined.

Response to A/LAI#8 Part 5, LRA Appendix C

"TLAAs are identified in Section 4. Fatigue usage calculations for the reactor vessel internals (lower core plate and control rod drive (CRD) guide tube pins) will consider the effects of the reactor water environment."

A.1.11 Fatigue Monitoring Program

"Fatigue usage calculations that consider the effects of the reactor water environment will be developed for a set of sample reactor coolant system 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 A.2.2.3."

B1.11 Fatigue Monitoring Program

"Fatigue usage calculations that consider the effects of the reactor water environment will be developed for a set of sample 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."

SQN LRA NRC Commitment change

Commitment #7.B, (See Enclosure 2, page E2-5)

"7.B. Fatigue usage calculations that consider the effects of the reactor water environment will be developed for a set of sample reactor coolant system 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."

RAI B.1.34-4

Background:

A/LAI No. 3 of MRP-227-A states, in part, that applicants/licensees of Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an existing program, or to identify changes to the programs that should be implemented to manage the aging of Westinghouse guide tube support pins. Section 3.2.5.3 of the staff's safety evaluation (SE), Rev.1 for MRP-227 clarifies, in part, that the evaluation consider the need to inspect the replacement Type 316 stainless steel support pins to ensure that cracking has been mitigated and that aging degradation is adequately monitored during the extended period of operation.

The applicant's response to A/LAI No. 3 in LRA Appendix C states that third generation split pins, which were qualified for 40 years from the time of installation, were installed in the fall of 2001 for Unit 1 and spring of 2002 for Unit 2. It further states that potential aging effects were evaluated, including those identified in MRP-191 Table 5-1, and no additional inspection requirements were established for the control rod guide tube support pins in the design change packages that installed them. LRA Appendix C states that the basis for not establishing additional inspection requirements is the following: (1) cold-worked Type 316 stainless steel split pins have been installed at other plants since 1997 and none of these plants have experienced any failures and (2) since other plants have installed split pins since 1997 and SQN did not install them until 2001 for Unit 1 and 2002 for Unit 2, the other plants will provide a leading indicator. Thus, the effects of aging on these components will be managed in the period of extended operation based on operating experience.

The Standard Review Plan for License Renewal (SRP-LR) Section A.1.2.3.4 states, in part, the effects of aging on a structure or component should be managed to ensure its availability to perform its intended function(s) as designed when called upon and that a program based solely on detecting structure and component failure should not be considered as an effective AMP for license renewal.

Issue:

The staff's SE, Rev.1, for MRP-227 specifically discusses the inspection of replacement Type 316 stainless steel support pins to ensure that cracking has been mitigated and that aging degradation is adequately monitored during the extended period of operation. Whereas, the applicant has stated that no additional inspection requirements were established for the control rod guide tube support pins and the effects of aging on these components will be managed in the period of extended operation based on operating experience from other plants.

The applicant's approach for aging management is not appropriate based on (1) the staff's SE, Rev. 1, for MRP-227, (2) A/LAI No. 3 and (3) SRP-LR Section A.1.2.3.4. It is not clear that it is appropriate for the applicant to rely solely on the operating experience at other plants as a means of aging management for its control rod guide tube support pins.

Request:

- *Justify that age-related degradation of the Type 316 stainless steel control rod guide tube support pins is adequately monitored during the extended period of operation in response to A/LAI No. 3 and Section 3.2.5.3 of the staff's SE, Rev.1 for MRP-227.*

  As part of the justification, provide the inspection category, techniques, frequency and coverage for the replacement Type 316 stainless steel control rod guide tube support pins to ensure that age-related degradation is adequately monitored during the extended period of operation.

  Revise the LRA, as needed, to provide program enhancements/augmentations.

TVA Response to RAI B.1.34-4

MRP-227-A, Section 4.4.3, states that guidance for the control rod guide tube (CRGT) support pins is limited to plant specific recommendations and that subsequent performance monitoring should follow the supplier recommendations.

For SQN Units 1 and 2, no additional performance monitoring recommendations were defined beyond the ASME Section XI program after completion of the Westinghouse-recommended replacement of the original Alloy X-750 CRGT support pins with the Type 316 stainless steel support pins. Consequently, age-related degradation of the SQN Unit 1 and Unit 2 CRGT support pins are managed during the period of extended operation by inspection performed in accordance with the ASME Section XI Program. The ASME Section XI Program provides the inspection category, techniques and frequency for the replacement CRGT support pins.

The Type 316 stainless steel CRGT support pins (also called split pins) were evaluated for irradiation-assisted stress corrosion cracking resistance, primary water stress corrosion cracking resistance, irradiation stress relaxation and creep, and embrittlement and toughness as a part of the design change governing their installation. As a result of this evaluation, no performance monitoring beyond the existing SQN ASME Section XI inspection program was required.

As stated in the response to Applicant/Licensee Action Item (A/LAI) # 3, the Type 316 stainless steel split pins were qualified for 40 years from the time of installation, which extends beyond the period of extended operation because the pins were installed in 2001 and 2002. As part of the ASME Section XI inspections described in LRA Section A.1.16, a VT-3 examination is performed on the accessible CRGT split pins each 10-year inservice inspection Interval.

The response to LRA Appendix C, A/LAI #3, page C-4 is with additions underlined and deletions shown with strikethrough.

"SQN installed the third generation of split pins in the fall of 2001 for Unit 1 and spring of 2002 for Unit 2. The new split pins were qualified for 40 years from the time of installation. Potential aging effects were evaluated, including those identified in MRP-191 Table 5-1. No additional inspection requirements were established for the control rod guide tube support pins in the design change packages that installed them based on the following.

- Cold-worked Type 316 SS split pins have been installed at other plants since 1997 and none of these plants have experienced any failures.

- Since other plants have installed split pins since 1997 and SQN did not install them until 2001 for Unit 1 and 2002 for Unit 2, the other plants will provide a leading indicator.

At SQN the effects of aging on ~~these components~~ the Type 316 stainless steel split pins will be managed in the period of extended operation ~~based on operating experience~~ using the ASME Section XI inspection requirements of the Inservice Inspection Program."

RAI B.1.34-5

Request:

- *Considering that the applicant's guide plates (cards) are fabricated of CASS, clarify the applicability of the MRP-191 and MRP-227-A that evaluated the guide plates (cards) as Type 304 stainless steel.*

- *Confirm that there are no other discrepancies in material fabrication of components evaluated in MRP-191 and MRP-227-A with those at the applicant's site.*

  If there are other discrepancies, provide the component (including material) and justify the aging effects and the inspection category, techniques, coverage, and frequency to account for the material differences.

  Revise the LRA and the response to A/LAI No. 2, as needed.
- *Since the guide plates (cards) are fabricated from CASS, describe and justify the plant-specific analysis performed in response to A/LAI No. 7 that considers the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and, if applicable, the limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques.*

- *Revise the response to A/LAI Nos. 2 and 7, as necessary.*

TVA Response to RAI B.1.34-5

TVA will respond to RAI B.1.34-5 by August 9, 2013. TVA discussed this extension with the NRC Project Manager, Mr. Richard Plasse in a telephone call on June 12, 2013 and he agreed verbally with this time extension.

RAI B.1.34-6

<u>Background:</u>

A/LAI No. 7 states, in part, the applicants/licensees of Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel, or precipitation hardened stainless steel materials.

A/LAI No. 7 continues to state that these analyses should also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. Furthermore, it states, in part, that this would apply to components fabricated from materials susceptible to thermal and/or irradiation embrittlement for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/License Action Item No. 2.

For Unit 2, the applicant stated in LRA Appendix C that the hold down spring is fabricated of Type 403 stainless steel, which is a martensitic stainless steel. Table 5-1 of MRP-191 also indicates that the Type 403 stainless steel hold down spring may be subject to thermal embrittlement.

<u>Issue:</u>

Since A/LAI No. 7 specifically discusses the performance of a plant-specific analysis for reactor vessel internal components fabricated from martensitic stainless steel materials, it is not clear whether the applicant has performed this analysis for the Unit 2 Type 403 hold down spring to consider the possible loss of fracture toughness due to thermal and irradiation embrittlement.

<u>Request:</u>

- *Clarify whether the Unit 2 Type 403 stainless steel hold down springs were evaluated in response to A/LAI No. 7.*
    - *If yes, describe and justify the evaluation performed to consider the possible loss of fracture toughness due to thermal and irradiation embrittlement.*
    - *If not, justify that the Unit 2 Type 403 stainless steel hold down spring is not applicable to the evaluation discussed in A/LAI No. 7.*

<u>TVA Response to RAI B.1.34-6</u>

The Unit 2 Type 403 stainless steel hold-down spring was not included in the evaluation discussed in the response to Applicant/License Action Item (A/LAI) #7 provided in LRA Appendix C.

Based on the Type 403 stainless steel low susceptibility to reduction in fracture toughness due to thermal and irradiation embrittlement, the Unit 2 Type 403 stainless steel hold-down spring will maintain its functionality under current licensing basis conditions as described in the response to RAI B.1.34-2.

The basis for this position is that an Expert panel, convened as part of the development of MRP-191, identified a low likelihood of failure of the Unit 2 Type 403 stainless steel hold-down springs and a low likelihood of damage resulting from a reduction in fracture toughness due to thermal or irradiation embrittlement.

The rationale for the decision of the expert elicitation is as follows. Thermal embrittlement of Type 403 stainless steel is not expected at the operating temperature of the pressurized water reactor (PWR). Irradiation embrittlement of Type 403 stainless steels can occur under higher accumulated neutron fluence. Because the hold-down spring is significantly above the active region of the core, the fluence to which the hold-down spring will be exposed over 60 years of operation is several orders of magnitude below the $1x10^{17}$ to $1x10^{18}$ n/cm^2 range of accumulated fluence that would be required to cause irradiation embrittlement in Type 403 stainless steels. At these low values of fluence, even combined thermal and irradiation effects are not expected to have any effect on the embrittlement behavior. Thus, the thermal and irradiation exposures, through the period of extended operation, will not result in any significant loss in fracture toughness for the Type 403 stainless steel hold-down spring.

Based on the results of the Expert elicitation, the Unit 2 Type 403 stainless steel hold-down spring was excluded from the list of those components that would require plant-specific functional assessments. This elicitation was documented in MRP-191 and was implicitly carried forward into the approach outlined in MRP-227-A.

As a result, Unit 2 Type 403 stainless steel hold-down springs are not within the scope of A/LAI No. 7 of MRP-227-A.

RAI B.1.34-7

Background:

During its audit, the staff noted that the "detection of aging effects" program element for the Reactor Vessel Internals Program indicates that the "Existing Programs" components were taken from Table 4-9 in MRP-227-A. Section 3.3 of MRP-227-A defines "Existing Programs" components as those PWR internals that generic and plant-specific existing aging management program elements are capable of managing aging effects. Section 4.4 of MRP-227-A states the following for "Existing Programs" components:

> *Included in the Existing Programs are PWR internals that are classified as removable core support structures. ASME Section XI, IWB-2500, Examination Category B-N-3 does not list component specific examination requirements for removable core support structures. Accordingly, factors such as original design, licensing and code of construction variability could result in significant differences in an individual plant's current B-N-3 requirements. These guidelines credit specific components contained within the general B-N-3 classification for maintaining functionality.*

Issue:

As an example, MRP-227-A noted that the "Existing Programs" components managed by ASME Section XI may vary from plant to plant based on original design, licensing and code of construction. Thus, since "Existing Programs" components may vary from plant to plant, it is not appropriate to rely on the list in Table 4-9 of MRP-227-A to determine the components that are classified as "Existing Programs" components. It is not clear whether the applicant confirmed during the Integrated Plant Assessment process when developing the LRA whether the components listed in Table 4-9 in MRP-227-A encompass all plant-specific "Existing Programs" components at Sequoyah Units 1 and 2.

In addition, during its review the staff noted that LRA Table 3.1.2-2 indicates the following:

- *Stainless steel "Interfacing components: Upper core plate alignment pins" is subject to cracking, which is managed by the Reactor Vessels Internals Program as an "existing programs" component*
- *Stainless steel "Control rod guide tube assembly and downcomer: Guide tube support pins (split pins)" is subject to cracking and loss of material – wear, which is managed by the Reactor Vessels Internals Program as an "existing programs" component.*

However, LRA Table C-3, "Existing Program Components at SQN Units 1 and 2" indicates that the "Alignment and interfacing components: Upper core plate alignment pins" are managed for loss of material (wear) by ASME Code Section XI but is silent about cracking. Furthermore, LRA Table C-3 does not identify the Control rod guide tube assembly and downcomer: Guide tube support pins (split pins) as an item or component that is managed by an existing program.

Request:

- *Clarify the discrepancies identified above between LRA Table 3.1.2-2 and LRA Table C-3 for the "Interfacing components: Upper core plate alignment pins" and "Control rod guide tube assembly and downcomer: Guide tube support pins (split pins)."*

  If revisions to the LRA are necessary, justify any revisions that are made.

- *Confirm that a review was performed to determine that the components in Table 4-9 of MRP-227-A encompass the plant-specific "Existing Programs" components at Sequoyah Units 1 and 2.*
  - *If not, justify that the sole use of Table 4-9 in MRP-227-A to determine the "Existing Programs" components is applicable to Sequoyah Units 1 and 2.*

TVA Response to RAI B.1.34-7

- The discrepancy related to the "Interfacing components: Upper core plate alignment pins" in LRA Table 3.1.2-2 and LRA Table C-3 exists because the Existing Program designator (**X**) associated with the cracking of the upper core plate alignment pins should have been Existing Program designator (**N**) or "No Additional Measures."

  It has been determined through the industry's efforts described in MRP-232 that no additional measures are necessary to address the aging effect of cracking. As described in MRP-232, the MRP determined that PWR chemistry is not conducive to cracking due to stress corrosion cracking (SCC). Chemistry parameters are controlled in accordance with the Water Chemistry Control – Primary and Secondary Program.

  Although gross cracking from SCC could be detected using the ASME Section XI, B-N-3, VT-3 inspections, the type of cracking caused by SCC in PWRs may not be detected via a VT-3 inspection. Therefore, no credit was taken in Table C-3 for the ASME Section XI inspections to detect cracking due to SCC for the SQN upper core plate alignment pins. The change to LRA Table 3.1.2-2 is to match Table C-3 by replacing the Existing Program designator (X) associated with cracking of the upper core plate alignment pins with (N) for "No Additional Measures."

  The Type 316 stainless steel "CRGT assembly and downcomer: Guide tube support pins (split pins)" were incorrectly classified as an Existing Program component in LRA Table 3.1.2-2. Both Type 316 stainless steel and Alloy X-750 guide tube support pins (split pins) were evaluated in MRP-191. Based on the screening process used in MRP-191, only the Alloy X-750 guide tube support pins were included in MRP-227-A, Table 3-3. Therefore the SQN Type 316 stainless steel guide tube support pins should be categorized as "No Additional Measures" (N) in LRA Table 3.1.2-2.
  The change to LRA Table 3.1.2-2, the component type "CRGT assembly and downcomer: Guide tube support pins (split pins)" line item is to designate the aging effects of cracking and loss of material due to wear with (N) rather than (X).

  Although the upper core plate alignment pins are not included in LRA Table C-3 for cracking, inspections for gross cracking of the upper core plate alignment pins are part of the Inservice Inspection Program. Similarly, the Type 316 stainless steel guide tube support pins were classified as Category A in MRP-191 and "No Additional Measures" in process Figure 2-2 of MRP-227-A, but are included in the ASME Section XI, B-N-3 inspections. The Inservice Inspection Program is described in LRA Section B.1.16 and is updated to the latest ASME Section XI code edition and addendum approved by the NRC in accordance with 10 CFR 50.55a.

- During the integrated plant assessment, a review of applicable aging effects from MRP-227-A, Table 3-3, Final Disposition of Westinghouse Components, and Table 4-9, Westinghouse Plants Existing Program Components, was performed to develop LRA Appendix C, Table C-3. It was determined that Table 4-9 included the SQN Units 1 and 2 Existing Program components. The paragraph from MRP-227-A, Section 4.4 quoted in the background of this RAI states that factors such as original design, licensing and code of construction variability could result in significant differences in an individual plant's current B-N-3 requirements. Table 4-9 does not include, nor is intended to include, all of the RVIs component inspection requirements of the Inservice Inspection Program. The process applied during the development of MRP-227-A, Table 4-9, identified components that have relevant aging effects that Existing Programs are credited to manage. Other components that are included in the examination category B-N-3, under the Inservice Inspection Program, are included in the primary, expansion, or "No Additional Measures" categories of MRP-227-A.

Enclosure 2

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

Regulatory Commitment List, Revision 2

(Only Commitment 7.B has been revised. See Page E2-5)

ENCLOSURE 2

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

Regulatory Commitment List, Revision 2

(Only commitment 7.B has been revised. See page E2-5)

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED 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.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.2
3	Implement the **Buried and Underground Piping and Tanks Inspection Program** as described in LRA Section B.1.4.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.4

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED 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	SOURCE	RELATED LRA SECTION / AUDIT ITEM
5	A. Revise **Diesel Fuel Monitoring Program** procedures to monitor and trend sediment and particulates in the standby DG day tanks. B. Revise Diesel Fuel Monitoring Program procedures to monitor and trend levels of microbiological organisms in the seven-day storage tanks. C. Revise Diesel Fuel Monitoring Program procedures to include a ten-year periodic cleaning and internal visual inspection of the standby DG diesel fuel oil day tanks and high pressure fire protection (HPFP) diesel fuel oil storage tank. These cleanings and internal inspections will be performed at least once during the ten-year period prior to the period of extended operation and at succeeding ten-year intervals. If visual inspection is not possible, a volumetric inspection will be performed. D. Revise Diesel Fuel Monitoring Program procedures to include a volumetric examination of affected areas of the diesel fuel oil tanks, if evidence of degradation is observed during visual inspection. The scope of this enhancement includes the standby DG seven-day fuel oil storage tanks, standby DG fuel oil day tanks, and HPFP diesel fuel oil storage tank and is applicable to the inspections performed during the ten-year period prior to the period of extended operation and succeeding ten-year intervals.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.8
6 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).	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.10

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

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
7	A. Revise **Fatigue Monitoring Program** procedures to monitor and track critical thermal and pressure transients for components that have been identified to have a fatigue Time Limited Aging Analysis. B. Fatigue usage calculations that consider the effects of the reactor water environment will be developed for a set of sample reactor coolant system 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. <u>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.</u> F_{en} factors will be determined as described in Section 4.3.3. C. Fatigue usage factors for the reactor coolant system limiting components will be determined to address the Cold Overpressure Mitigation System (COMS) event (i.e., low temperature overpressurization event) and the effects of the structural weld overlays. D. Revise Fatigue Monitoring Program procedures to provide updates of the fatigue usage calculations 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.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.11
8	A. Revise **Fire Protection Program** procedures to include an inspection of fire barrier walls, ceilings, and floors for any signs of degradation such as cracking, spalling, or loss of material caused by freeze thaw, chemical attack, or reaction with aggregates. B. Revise Fire Protection Program procedures to provide acceptance criteria of no significant indications of concrete cracking, spalling, and loss of material of fire barrier walls, ceilings, and floors and in other fire barrier materials.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.12
9	A. Revise Fire Water System Program procedures to	SQN1: Prior to		B.1.13

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
9	include periodic visual inspection of fire water system internals for evidence of corrosion and loss of wall thickness. B. Revise Fire Water System Program procedures to include one of the following options: • Wall thickness evaluations of fire protection piping using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material will be performed prior to the period of extended operation and periodically thereafter. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function. • A visual inspection of the internal surface of fire protection piping will be performed upon each entry into the system for routine or corrective maintenance. These inspections will be capable of evaluating (1) wall thickness to ensure against catastrophic failure and (2) the inner diameter of the piping as it applies to the design flow of the fire protection system. Maintenance history shall be used to demonstrate that such inspections have been performed on a representative number of locations prior to the period of extended operation. A representative number is 20% of the population (defined as locations having the same material, environment, and aging effect combination) with a maximum of 25 locations. Additional inspections will be performed as needed to obtain this representative sample prior to the period of extended operation and periodically during the period of extended operation based on the findings from the inspections performed prior to the period of extended operation. C. Revise Fire Water System Program procedures to ensure a representative sample of sprinkler heads will be tested or replaced before the end of the 50-year sprinkler head service life and at ten-year intervals thereafter during the extended period of operation. NFPA-25 defines a representative sample of sprinklers to consist of a minimum of not less than four sprinklers or one percent of the number of sprinklers per individual sprinkler sample, whichever is greater. If the option to replace the sprinklers is chosen, all sprinkler heads that have been in service	09/17/20 SQN2: Prior to 09/15/21		

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
(cont.)	for 50 years will be replaced. D. Revise Fire Water System Program procedures to consider implementing the flow testing requirements of NFPA 25 or justify why the flow testing requirements of NFPA should not be implemented. E. Revise Fire Water System Program procedures to include acceptance criteria for periodic visual inspection of fire water system internals for corrosion, minimum wall thickness, and the absence of biofouling in the sprinkler system that could cause corrosion in the sprinklers.			
10	Revise **Flow Accelerated Corrosion Program** procedures to implement NSAC-202L guidance for examination of components upstream of piping surfaces where significant wear is detected.	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	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.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.17
13 13 (cont.)	Inspection of **Overhead Heavy Load and Light Load** (Related to Refueling) **Handling Systems**: A. Revise program procedures to specify the inspection scope will include monitoring of rails in the rail system for wear; monitoring structural components of the bridge, trolley and hoists for the aging effect of deformation, cracking, and loss of material due to corrosion; and monitoring structural connections/bolting for loose or missing bolts, nuts, pins or rivets and any other conditions indicative of loss of bolting integrity. B. Revise program procedures to include the inspection and inspection frequency requirements of	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.18

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
	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 period of extended operation and at least every ten years thereafter based on initial testing to determine possible changes in boron-10 areal density. B. Revise Neutron Absorbing Material Monitoring Program procedures to relate physical measurements of Boral coupons to the need to perform additional testing. C. Revise Neutron Absorbing Material Monitoring Program procedures to perform trending of coupon testing results to determine the rate of degradation and to take action as needed to maintain the intended function of the Boral.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.22
17	Implement the **Non-EQ Cable Connections Program** as described in LRA Section B.1.24	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.24
18	Implement the **Non-EQ Inaccessible Power Cable** (400 V to 35 kV) **Program** as described in LRA Section B.1.25	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.25

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
19	Implement the **Non-EQ Instrumentation Circuits Test Review Program** as described in LRA Section B.1.26.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.26
20	Implement the **Non-EQ Insulated Cables and Connections Program** as described in LRA Section B.1.27	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.27
21	A. Revise **Oil Analysis Program** procedures to monitor and maintain contaminants in the 161-kV oil filled cable system within acceptable limits through periodic sampling in accordance with industry standards, manufacturer's recommendations and plant-specific operating experience. B. Revise Oil Analysis Program procedures to trend oil contaminant levels and initiate a problem evaluation report if contaminants exceed alert levels or limits in the 161-kV oil-filled cable system.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.28
22	Implement the **One-Time Inspection Program** as described in LRA Section B.1.29.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.29
23	Implement the **One-Time Inspection – Small Bore Piping Program** as described in LRA Section B.1.30	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.30
24	Revise **Periodic Surveillance and Preventive Maintenance Program** procedures as necessary to include all activities described in the table provided in the LRA Section B.1.31 program description.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.31
25	A. Revise **Protective Coating Program** procedures to clarify that detection of aging effects will include inspection of coatings near sumps or screens associated with the emergency core cooling system. B. Revise Protective Coating Program procedures to clarify that instruments and equipment needed for inspection may include, but not be limited to, flashlights, spotlights, marker pen, mirror, measuring tape, magnifier, binoculars, camera with or without wide-angle lens, and self-sealing polyethylene sample bags.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.32

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
	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 take physical measurements of the Type 304 stainless steel hold-down springs in Unit 1 at each refueling outage to ensure preload is adequate for continued operation. B. Revise Reactor Vessel Internals Program procedures to include preload acceptance criteria for the Type 304 stainless steel hold-down springs in Unit 1.	SQN1: Prior to 09/17/20 SQN2: Not Applicable		B.1.34
28	A. Revise **Reactor Vessel Surveillance Program** procedures to consider the area outside the beltline such as nozzles, penetrations and discontinuities to determine if more restrictive pressure-temperature limits are required than would be determined by just considering the reactor vessel beltline materials. B. Revise Reactor Vessel Surveillance Program procedures to develop an NRC-approved schedule for capsule withdrawals to meet ASTM-E185-82 requirements, including the possibility of operation beyond 60 years. C. Revise Reactor Vessel Surveillance Program procedures to withdraw and test a standby capsule to cover the peak fluence expected at the end of the period of extended operation.	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.35
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

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
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 31 (cont.)	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;	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.40

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
31 (cont.)	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 • 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) • 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)			

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
31 (cont.)	• 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 • 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			

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
31 (cont.)	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 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			

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
31 (cont.)	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). 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. • 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.			

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
31 (cont.)	I. Verify acceptance criteria in Structures Monitoring Program procedures is based on information provided in industry codes, standards, and guidelines including NEI 96-03, ACI 201.1R-92, ANSI/ASCE 11-99 and ACI 349.3R-02. Industry and plant-specific operating experience will also be considered in the development of the acceptance criteria.			
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	SQN1: Prior to 09/17/20 SQN2: Prior to 09/15/21		B.1.42

No.	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
33 (cont.)	additional analysis. E. Revise Water Chemistry Control - Closed Treated Water Systems Program procedures to inspect a representative sample of piping and components at a frequency of once every ten years for the following systems: • Standby diesel generator jacket water subsystem • Component cooling system • Glycol cooling loop system • High pressure fire protection diesel jacket water system • Chilled water portion of miscellaneous HVAC systems (i.e., auxiliary building, Incore Chiller 1A, 1B, 2A, & 2B, and 6.9 kV Shutdown Board Room A & B) F. Components inspected will be those with the highest likelihood of corrosion or cracking. A representative sample is 20% of the population (defined as components having the same material, environment, and aging effect combination) with a maximum of 25 components. These inspections will be in accordance with applicable ASME Code requirements, industry standards, or other plant-specific inspection and personnel qualification procedures that ensure the capability of detecting corrosion or cracking.			

The above table identifies the 33 SQN NRC LR commitments. Any other statements in this letter are provided for information purposes and are not considered to be regulatory commitments.