TVA

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

July 30, 2013

10 CFR Part 54

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

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

Subject: **Revised Response to NRC Request for Additional Information Regarding the Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application, Sets 1 and 3 (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. TVA Letter to NRC, "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," dated June 25, 2013 (ADAMS Accession No. ML13178A283)
3. TVA Letter to NRC, "Response to NRC Request for Additional Information Regarding the Scoping and Screening Methodology Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application, Set 3," dated June 7, 2013 (ADAMS Accession No. ML13163A442)

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 Reference 2, TVA submitted a response to the NRC's request for additional information (RAI) Set 1, which included RAIs B.1.34-1 and B.1.34-6.

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In a July 10, 2013 telecom, the NRC License Renewal Project Manager, Mr. Richard Plasse, requested clarification with regard to these two questions. Enclosure 1 provides the revision to RAIs B.1.34-1 and B.1.34-6 providing the requested clarification.

In addition, by Reference 3, TVA submitted a response to the NRC's RAI Set 3, which included RAI 2.1-1. TVA has revised the response to this RAI for clarity and has also included it in the Enclosure.

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

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

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

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosure: Revised Responses to RAI Questions B.1.34-1, B.1.34-6, and 2.1-1

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

ENCLOSURE

Tennessee Valley Authority

Sequoyah Nuclear Plant, Units 1 and 2 License Renewal

Revised Responses to RAIs B.1.34-1, B.1.34-6, and 2.1-1

NRC Set 1, 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 Set 1, RAI B.1.34-1 (Revised)

- 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 assumed linear decrease in height with time is conservative compared to the expected exponential decay over time for pure stress relation. Additional margin may be obtained by considering an exponential decay using the basis method for a stress-creep model per WCAP-17096.

  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 assumed linear decrease in height with *time is conservative compared to the expected exponential decay over time for pure* stress relation. Additional margin may be obtained by considering an exponential decay using the basis method for a stress-creep model per WCAP-17096.

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. The 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."

NRC Set 1, RAI B.1.34-6

Background:

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/LA I 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 A/LAI 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.

Issue:

Since A/LAI No. 7 specifically discusses the performance of a plant-specific analysis for RVI 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.

Request:

- *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.*

TVA Response to Set 1, RAI B.1.34-6 (Revised)

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 Type 403 stainless steel hold-down springs and a low likelihood of damage resulting from reduction in fracture toughness due to thermal or irradiation embrittlement.

The rationale for the decision of the Expert elicitation is as follows.

1. Failure resulting from thermal embrittlement of Type 403 stainless steel is not expected at the operating temperature of the pressurized water reactor (PWR). MRP-191 considered the potential for aging degradation due to thermal embrittlement of Type 403 hold-down springs and concluded that, although Type 403 stainless steel must be considered susceptible to thermal embrittlement, the likelihood of failure was low. Considerations of the thermal embrittlement of Type 403 and Type 410 stainless steels had been previously assessed in MRP-80 and MRP-175. (Type 403 is a more tightly controlled version of Type 410, sometimes referred to as "Turbine Grade".) MRP-80 and MRP-175 noted that, while thermal embrittlement of Types 403 and 410 stainless steels was possible, no failures of Types 403 and 410 stainless steels in operating plant equipment had ever been attributed to thermal embrittlement. Laboratory-induced thermal embrittlement of Type 410 stainless steel was reported, but this was produced by accelerated kinetics (i.e., higher than operational temperatures). Thus, the rates of embrittlement of Type 410 and Type 403 stainless steels are expected to be very low at operating temperatures. Moreover, these data were reported for Type 410; Type 403 stainless steel contains lower levels of silicon and phosphorous. Therefore, Type 403 stainless steel is expected to be even less susceptible to thermal embrittlement than Type 410 stainless steel. On this basis, the likelihood of failure by thermal embrittlement of Type 403 hold-down springs was considered to be "Low" in MRP-191.

2. Irradiation embrittlement of Type 403 stainless steels can occur under higher accumulated neutron fluence. Because the hold-down spring is significantly removed from the core, the fluence to which the hold-down spring is exposed over 60 years of operation is several orders of magnitude below the 1×10^{17} to 1×10^{18} n/cm^2 range of accumulated fluence that would be required to cause irradiation embrittlement in ferritic and martensitic steels. At these low values of fluence, even combined thermal and irradiation effects are not expected to have any effect in embrittling the Type 403 stainless steel hold-down spring material.

Based on the results of the Expert elicitation, irradiation embrittlement was not considered a viable mechanism of aging degradation. Though thermal embrittlement was considered a viable mechanism of degradation, this mechanism was already considered in MRP-191. Since there is no additional mechanism of degradation, the Expert elicitation concluded that the Unit 2 Type 403 stainless steel hold-down spring would be excluded from the list of those components that would require plant-specific functional assessments. This Expert elicitation process was documented in MRP-191 and was implicitly carried forward into the approach outlined in MRP-227-A. As a result, Type 403 stainless steel hold-down springs are not within the scope of A/LAI No. 7 of MRP-227-A.

NRC Set 3, RAI 2.1-1

Background:

Title 10 of the Code of Federal Regulations (CFR) 54.4, "Scope," states, in part:

(a) Plant systems, structures and components [SSCs] within the scope of this part are –

(1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions –

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.

Issue:

During the on-site scoping and screening methodology audit, the staff determined that the applicant had used a plant equipment database, which provides the component quality classification, as an information source used in identifying SSCs within the scope of license renewal. The plant equipment database uses the terms "safety-related" or "SR" to identify safety-related SSCs. However, during the audit the staff determined that not all components identified as safety-related in the plant equipment database were included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1).

Request:

The staff requests that the applicant perform a review of this issue and provide a description of the process used to evaluate components identified as safety-related in the plant equipment database and the basis for not including components identified as safety-related within the scope of license renewal in accordance with 10 CFR 54.4(a)(1). Indicate if the review concludes that use of the scoping methodology precluded the identification of SSCs that should have included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1).

Describe any additional scoping evaluations performed to address the 10 CFR 54.4(a)(1) criteria. List any additional SSCs included within the scope of license renewal as a result of the review, and any structures and components (SCs) for which aging management reviews were performed.

TVA Response to Set 3, RAI 2.1-1

As described in the License Renewal Application (LRA) Sections 2.1.1 and 2.1.2, mechanical components identified as safety-related in the plant equipment database are included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1). These components are subject to aging management review if they are not subject to replacement based on a qualified life or specified time period, and they perform an intended function, as described in § 54.4, without moving parts or without a change in configuration or properties.

Although some structural components and many electrical components are identified in the plant equipment database as safety-related, neither structural nor electrical scoping and screening use the plant equipment database. See LRA Sections 2.1.1, 2.1.2.2 and 2.1.2.3 for descriptions of structural and electrical scoping and screening.

In response to the NRC request, mechanical components classified as safety-related in the plant equipment database but determined not to be subject to an aging management review were re-evaluated.

The process used in this re-evaluation was to confirm that these mechanical components are either:

(1) Subject to replacement based on a qualified life or specified time period,

(2) Perform their intended function with moving parts or a change in configuration or properties, or

(3) Do not perform a safety function as defined in 10 CFR 54.4(a)(1).

The review, following the process described above, confirmed that the mechanical components classified as safety-related in the plant equipment database and determined <u>not</u> to be subject to aging management review either

1) Are replaced based on a qualified life or specified time period, or

2) Perform their intended function with moving parts or a change in configuration or properties, or

3) Do not perform a license renewal intended function defined in 10 CFR 54.4(a)(1), but had been conservatively classified as safety-related based on management decision.

The review concluded that the use of the scoping methodology did not preclude the identification of systems, structures, or components (SSCs) that should have been included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1).

No additional scoping evaluations were necessary to address the 10 CFR 54.4(a)(1) criteria, no additional SSCs were included within the scope of license renewal as a result of this review and no additional aging management reviews were necessary.

<u>Previous Response to Set 3, RAI 2.1-1, 5th Paragraph:</u>

The review, following the process described above, confirmed that the mechanical components classified as safety-related in the plant equipment database and determined <u>not</u> to be subject to aging management review are either <u>*being replaced based on a qualified life or specified time period, perform their intended function with moving parts or a change in configuration or properties. These components are conservatively classified as safety-related based on management decision but have no safety function as defined in 10 CFR 54.4(a)(1).*</u>

Revised Response to Set 3, RAI 2.1-1, 5th Paragraph:

The review, following the process described above, confirmed that the mechanical components classified as safety-related in the plant equipment database and determined not to be subject to aging management review either

1) Are replaced based on a qualified life or specified time period, or

2) Perform their intended function with moving parts or a change in configuration or properties, or

3) Do not perform a license renewal intended function defined in 10 CFR 54.4(a)(1), but had been conservatively classified as safety-related based on management decision.