



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 18, 2014

LICENSEE: Tennessee Valley Authority

FACILITY: Sequoyah Nuclear Plant, Units 1 and 2

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON JUNE 18, 2013, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND TENNESSEE VALLEY AUTHORITY, CONCERNING REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC. NOS. MF0481 AND MF0482) – SET 8.

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of Tennessee Valley Authority held a telephone conference call on June 18, 2013, to discuss and clarify the staff's requests for additional information (RAIs) concerning the Sequoyah Nuclear Plant, Units 1 and 2, license renewal application. The telephone conference call was useful in clarifying the intent of the staff's RAIs.

Enclosure 1 provides a listing of the participants and Enclosure 2 contains a listing of the RAIs discussed with the applicant, including a brief description on the status of the items.

The applicant had an opportunity to comment on this summary.

A handwritten signature in black ink, appearing to read "R. Plasse" followed by "FOR" in a smaller, less distinct script.

Richard A. Plasse, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:

1. List of Participants
2. List of Requests for Additional Information

cc: Listserv

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/RA by Emmanuel Sayoc for/

Richard A. Plasse, Project Manager
Projects Branch 1
Division of License Renewal
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ADAMS Accession No.: **ML13172A012**

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NAME	Y Edmonds	E Sayoc	Y Diaz-Sanabria	R Plasse (<i>ESayoc for</i>)
DATE	1/24/14	1/24/14	2/14/14	2/18/14

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Letter to J. Shea from R. Plasse dated February 18, 2014

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON JUNE 18, 2013, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND TENNESSEE VALLEY AUTHORITY, CONCERNING REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC. NOS. MF0481 AND MF0482) – SET 8.

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TELEPHONE CONFERENCE CALL
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION

LIST OF PARTICIPANTS

JUNE 18, 2013

PARTICIPANTS

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Emmanuel Sayoc
James Medoff
Alice Erickson
Henry Lee
Dennis Lundy

AFFILIATIONS

U.S. Nuclear Regulatory Commission (NRC)
NRC
NRC
NRC
Tennessee Valley Authority (TVA)
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REQUESTS FOR ADDITIONAL INFORMATION DISCUSSED
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
JUNE 18, 2013

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of Tennessee Valley Authority held a telephone conference call on June 18, 2013, to discuss and clarify the following requests for additional information (RAIs) concerning the license renewal application (LRA).

RAI 4.3.1-1 - Based on the discussion with the applicant, the staff agreed to revise this RAI as follows. This RAI was sent to the applicant in letter dated June 24, 2013 (ML13150A412).

Background:

Technical Specification (TS) 6.8.4.1 provides controls to track the updated final safety analysis report (UFSAR) Section 5.2.1 cyclic and transient occurrences to ensure that components are maintained within the design limit, UFSAR Table 5.2.1-1 identifies the reactor coolant system (RCS) design transients,

Issue:

1. License renewal application (LRA) Tables 4.3-1 and 4.3-2 list pressurizer heatups as a normal operating condition transient, but this transient is not defined as a design transient for normal operating conditions in UFSAR Table 5.2.1-1 for the RCS.
2. UFSAR Table 5.2.1-1 lists the 10percent step load increase and decrease transient as an applicable design basis transients; however, these transients are not listed as transients that would need to be monitored in LRA Tables 4.3-1 and 4.3-2. It is not evident why the Fatigue Monitoring Program would not need to monitor the 10percent step load increase and decrease normal operating condition transient, as this would be required to be performed in accordance with the appropriate TS requirements.
3. LRA Tables 4.3-1 and 4.3-2 list 10 cycles as the design cycle limit for the pressurizer auxiliary spray actuations transient; however, UFSAR Table 5.2.1-1 identifies that the cycle limit for this transient is 12 cycles.

Request:

1. Provide the basis why UFSAR Table 5.2.1-1 does not list pressurizer heatups as an applicable normal operating condition transient, when this transient is listed in LRA Tables 4.3-1 and 4.3-2. Clarify and justify whether a 10 CFR 50.71(e) update of UFSAR Table 5.2.1-1 will need to be processed to add the pressurizer heatup transient as a normal operating condition transient for the Safety Class 1 or Class A components at the units.
2. Provide the basis why the Fatigue Monitoring Program would not need to monitor the 10 step load increase and decrease normal operating condition. Specifically, justify why the monitoring of these transients would not need to be performed in accordance with the applicable TS 6.8.4.1 requirements for the units.
3. Justify the basis for reporting a different value for the cycle limit for the pressurizer auxiliary spray actuations transient (i.e., 12 cycles) in UFSAR Table 5.2.1-1 that is

different from the cycle limit for this transient in LRA Tables 4.3-1 and 4.3-2 (i.e., 10 cycles).

RAI 4.3.2-1 -Based on the discussion with the applicant, the staff agreed to revise this RAI as follows. This RAI was sent to the applicant in letter dated June 24, 2013 (ML13150A412).

Background:

LRA Section 4.3.2 discusses the maximum allowable stress range reduction analyses (implicit fatigue analyses for those Non-Safety Class 1 or Non Safety Class A piping systems that were designed either to the USAS B31.1 design code or those in the ASME Code Section III requirements for Class 2 or 3 components. These implicit fatigue analyses are identified as time-limited aging analyses (TLAAs) for the LRA. The evaluation of the TLAAs in accordance with 10 CFR 54.21(c)(1)(i) is conducted by comparing the cumulative number of full thermal range transient occurrences for the components to a value of 7000 cycles in order to demonstrate that the maximum allowable stress ranges for the components would not need to be reduced.

Issue:

LRA Section 4.3.2 does not identify which Non-Safety Class 1 or Non-Safety Class A piping systems in the engineered safety feature (ESF) systems, auxiliary (AUX) systems, or steam and power conversion (SPC) systems were within the scope of the applicable implicit fatigue analysis requirements in either the USAS B31.1 design code or in the ASME Section III provisions for Class 2 or 3 components.

Also, LRA Section 4.3.2 does not identify the type of piping components and piping elements that are within the scope of these analyses or identify which design transients are characterized as full thermal range transients for the implicit fatigue analyses of these Non-Safety Class 1 or Non-Safety Class A piping components and elements.

Request:

1. Identify all Non-Safety Class 1 or Non-Safety Class A ESF, AUX, and SPC systems, and the piping components and elements in these systems, that are within the scope of the applicable implicit fatigue analysis requirements in the USAS B31.1 design code or the ASME Code Section III provisions for Class 2 or 3 components. For these systems, identify the design basis transients that constitute "full thermal range" transients for the implicit fatigue analysis of these non-Class 1/non-Class A systems. Justify that the total number of the cycles for the "full thermal range" transients will remain less than or equal to the limit of 7000 cycles during the period of extended operation.
2. Compare the systems and components in the response to Part a. of this (RAI) to the list of components in the "Table 2" AMR tables for those ESF, AUX, and SPC systems. Amend the LRA accordingly if it is determined that additional aging management review (AMR) items on "cracking – fatigue" need to be identified for the LRA's AMR tables for ESF, AUX, and SPC systems.

3. Revise LRA Appendix A as appropriate based on the response.

RAI 4.3.3-1 -Based on the discussion with the applicant, the staff agreed to revise this RAI as follows. This RAI was sent to the applicant in letter dated June 24, 2013 (ML13150A412).

Background:

LRA Section 4.3.3 provides the applicant's environmentally-assisted fatigue evaluations for Safety Class 1 or Safety Class A locations in the reactor coolant pressure boundary. The applicant provides its environmentally-assisted fatigue results (i.e., F_{en} -adjusted cumulative usage factor results or $CU-F_{en}$ results) for these components in LRA Table 4.3-12. The applicant identifies that the $CU-F_{en}$ results were calculated using the recommended formulas in NUREG/CR-6583 for carbon or low-alloy steel components and in NUREG/CR-5704 for those stainless steel components.

Issue:

The F_{en} values that are derived in accordance with the NUREG report formulas are dependent on plant parameter inputs, such sulfur content and dissolved oxygen impurity contents for the reactor coolant, the operating temperature of the coolant, and strain-rates for the components. It is not evident to the staff which plant parameter assumptions were used to establish the F_{en} value of 2.45 for Safety Class 1/Safety Class A components made from low alloy steel or carbon steel materials or the F_{en} value of 15.36 for Safety Class 1/Safety Class A components made from stainless steel materials.

Request:

Clarify how the F_{en} values for the low-alloy steel or carbon steel components and for stainless steel components were derived in accordance with applicable NUREG methodology for the respective material type. Identify and justify any assumptions on the plant parameter inputs (e.g., sulfur content, temperature, dissolved oxygen, and strain rate parameters) that were used to derive the F_{en} factors for these Safety Class 1/Safety Class A components.

RAI 3.3.2.3.13-1 – This RAI was deleted as it was a duplicate of an already issued RAI with the same number in letter dated June 21, 2013 (ML13144A734).

RAI 3.1.2.2.1-1 – Discussed however no changes were made and a mutual understanding was reached by the staff and the applicant. This RAI was sent to the applicant in letter dated June 24, 2013 (ML13150A412).

RAI B.1.40-6 - Based on the discussion with the applicant, the staff agreed to revise this RAI as follows. This RAI was sent to the applicant in letter dated June 24, 2013 (ML13150A412).

Background:

LRA Section B.1.40, Structures Monitoring, states an enhancement to the “scope of program” program element. In this enhancement, the applicant stated that the Structures Monitoring Program procedures will be revised to specify each of the in-scope structures and structural components for each of the Structures Monitoring, Regulatory Guide (RG) 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants, and Masonry Wall AMPs.

Issue:

The staff understands this revision to the Structures Monitoring Program procedures to be an enhancement (Commitment 31) to the existing program in order to make the program consistent with the GALL Report; however, because this is an existing program, it is not clear which structures and structural components are being added to the scope of the program for license renewal, that are not already within the existing program.

Request:

Identify the structures and structural components and commodities that are being added to the scope of the Structures Monitoring Program for license renewal, that are not currently listed in the existing Structures Monitoring Program.

RAI B.1.40-7 – discussed but no changes were made and a mutual understanding was reached by the staff and the applicant. This RAI was sent to the applicant in letter dated June 24, 2013 (ADAMS Accession Number ML13150A412).

RAI 3.6-3 -Based on the discussion with the applicant, the staff agreed to revise this RAI as follows. This RAI was sent to the applicant in letter dated June 24, 2013 (ML13150A412).

Background:

In LRA Section 3.6.2.2.3, the applicant states that the design of switchyard bus bolted connections precludes torque relaxation as confirmed by plant-specific operating experience. The design of switchyard bolted connections includes Belleville washers. The type of bolting plate and the use of Belleville washers is the industry standard to preclude torque relaxation.

Issue:

EPRI document TR-104213, “Bolted Joint Maintenance & Application Guide,” identifies a special problem with Belleville washers. It states that hydrogen embrittlement is a recurring problem with Belleville washers and other springs. When springs are electroplated, the plating process forces hydrogen into the metal grain boundaries. If the hydrogen is not removed, the spring may spontaneously fail at any time while in service.

Request:

Identify if electroplated Belleville washers are currently used at SQN. If they are, explain why hydrogen embrittlement is not a concern for switchyard bus bolted connections at SQN.

RAI 4.7.2-2 -Based on the discussion with the applicant, the staff agreed to revise this RAI as follows. This RAI was sent to the applicant in letter dated June 24, 2013 (ML13150A412).

Background:

LRA Section 4.7.2 states that “No other cranes [besides the manipulator crane] at SQN were built to CMAA-70 requirements...The SQN responses to NUREG-0612 and the review of the site cranes identified that the reactor building polar crane and the auxiliary building crane were not built to the structural fatigue requirements of CMAA-70.”

UFSAR Section 3.12.4.1 contains SQN commitments in response to NUREG 0612, which recommends compliance with seven guidelines to ensure the Control of Heavy Loads Program is adequate. Guideline 7 states that “The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of B301.2-1976 and CMAA-70.” The applicant’s response states “The actual design data for the auxiliary building crane and the reactor building crane were compared with the guidelines of CMAA-70 and ANSI (ASME) B30.2.” Where specific compliance was not evident by review, an evaluation was made by imposing these guidelines on the actual design...this was the approach used for evaluating the design of major structural components by using load combinations and allowable stresses given in CMAA-70. The results of this review and analysis indicate that both cranes meet or exceed the requirements of CMAA-70 and ANSI (ASME) B30.2.”

UFSAR Section 3.8.6.2.2 “Applicable Codes, Standards, and Specifications” of the SQN UFSAR states that the requirements of CMAA-70 were used to upgrade the Auxiliary Building Crane to single failure proof crane systems.

Issue:

While the original design of the reactor building and auxiliary building cranes may not have directly incorporated the guidelines of CMAA-70 and ANSI B30.2, several analyses have been done to compare the design of the auxiliary building and reactor building cranes to CMAA-70 and ANSI B30.2 to demonstrate compliance with the guidance outlined in NUREG 0612. In addition, CMAA-70 was used in an analysis to upgrade the Auxiliary Building Crane design. The staff believes that since these analyses and comparisons to the criteria and guidelines of CMAA-70 and ANSI B30.2 for the auxiliary building crane and reactor building crane are outlined in the UFSAR, the applicant’s review meets the criteria for a TLAA.

Request:

Provide basis for the conclusion that current licensing basis does not incorporate the applicable design specifications of CMAA-70 and ANSI B30.2, and does not consider the TLAA analyses for the Auxiliary and Reactor Building Cranes.