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

CNL-14-018

February 13, 2014

10 CFR 50.4
10 CFR 50.90

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 Sequoyah Nuclear Plant (SQN), Units 1 and 2 - Application to Modify Ice Condenser Technical Specifications to Address Revisions in Westinghouse Mass and Energy Release Calculation (SQN-TS-12-04)(TAC Nos. MF2446 and MF2447)**

References:

1. Letter from TVA to NRC, "Application to Modify Ice Condenser Technical Specifications to Address Revisions in Westinghouse Mass and Energy Release Calculation (SQN-TS-12-04)," dated July 3, 2013, (ADAMS Accession Number ML13199A281)

2. Letter from NRC to TVA, "Sequoyah 1 and 2 - RAIs for License Amendment Request Regarding Modification of Ice Condenser Technical Specifications to Address Revisions in Westinghouse Mass and Energy Release Calculation (SQN-TS-12-04)," dated January 14, 2014 (ADAMS Accession No. ML14016A281)

By letter dated July 3, 2013 (Reference 1), Tennessee Valley Authority (TVA) submitted a request for an amendment to Facility Operating License Nos. DPR-77 and DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. The license amendment request (LAR) proposed changes to revise SQN, Units 1 and 2, Technical Specifications (TSs) 3/4.6.5, "Ice Condenser," to increase the total ice weight from 2,225,880 pounds to 2,540,808 pounds. This change is necessary to address the issues raised in Nuclear Safety Advisory Letter (NSAL) 11-5, "Westinghouse LOCA Mass and Energy Release Calculation Issues."

The Enclosure to this letter provides TVA's RAI responses. There are no changes required to the LAR as submitted in the Reference 1 letter as a result of this additional information. 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 amendment previously provided in Reference 1. TVA has further determined that the proposed amendment still qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the enclosure to the Tennessee Department of Environment and Conservation.

There are no new regulatory commitments included in this submittal. Please address any questions regarding this submittal to Mr. Henry Lee at (423) 843-4104.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 13th day of February 2014.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosure:
Response to Request for Additional Information

cc (Enclosure):

NRC Regional Administrator - Region II
NRC Resident Inspector – Sequoyah Nuclear Plant
Director, Division of Radiological Health, Tennessee State Department of Environment and Conservation

CWS/EDS
Enclosure
bcc (Enclosure)

NRC Project Manager – Sequoyah Nuclear Plant
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Enclosure

Tennessee Valley Authority (TVA)
Sequoyah Nuclear Plant, Units 1 and 2

Response to Request for Additional Information

RAI-1

Reference 1, Attachment 5 to Enclosure, Section 1.0 states:

> *"In addition, for a comprehensive reconciliation of all issues relative to the LOCA mass and energy release analysis of record (AOR) all appropriate corrections relative to NSAL-06-6 (Reference 10) were also addressed."*

Provide a discussion of issues described in NSAL-06-6 that are resolved in the containment integrity reanalysis documented in Attachment 5.

TVA RESPONSE:

In NSAL-11-5 (Reference 2) and NSAL-06-6 (Reference 3), Westinghouse documented several issues with past guidance provided for the performance of loss-of-coolant accident (LOCA) mass and energy (M&E) release analysis (References 5 and 6).

Reference 3 documented eight issues that could potentially affect the M&E results used to evaluate containment integrity. Westinghouse determined that the current analysis (Reference 4) for the Sequoyah Nuclear Plant (SQN), Units 1 and 2, at the time when NSAL-06-6 was issued was affected by a subset of these issues. The results of a reanalysis showed that there was no adverse effect on the calculated peak containment pressure or initial ice mass due to the eight issues and that the current analysis remained valid. No changes to the existing computer programs were needed.

The eight issues described in NSAL-06-6 that could potentially affect the M&E results are as follows:

<u>1) Area of the Downcomer in the REFLOOD Code</u>

Westinghouse-designed reactors can be divided into downflow and upflow barrel baffle designs. The original guidance for calculating the downcomer area for downflow plants was incorrect. When the calculation was corrected, a larger downcomer area and thus a larger volume for downflow plants was calculated. This resulted in a longer time required for the emergency core cooling system (ECCS) to completely fill the downcomer. This issue affected the current analysis and was corrected in the proposed analysis.

<u>2) Area of the Upper Plenum in the FROTH Code</u>

The FROTH computer program is run in conjunction with the REFLOOD computer program and calculates the LOCA M&E releases for the post-reflood period until the steam generator (SG) secondary side pressure(s) is calculated to equilibrate at the containment design pressure.

During this time period, the two-phase mixture levels in the core, upper plenum, hot leg and SG inlet plenum are the principle parameters of interest. Due to a misinterpretation of a database parameter, the cross-sectional area of the upper plenum (AUPP) was being over predicted, which resulted in a reduction in entrainment to the SGs and thus less steam production. This issue affected the current analysis and was corrected in the proposed analysis.

3) Review of Other FROTH Inputs

A review of the FROTH code user controlled input variables showed that the variable for the SG inlet plenum area (i.e., variable ASGP), which is used to calculate the void fraction in the SG inlet plenum, was based on a value that was generally too small. A review of SG geometry for the inlet plena determined a more appropriate method for calculating ASGP. The result was a larger flow area and a reduction in entrained liquid. The reduction in entrained liquid reduces the mass and energy released post-reflood and is a benefit for the calculated containment pressure. This issue affected the current analysis and was corrected in the proposed analysis.

4) WCAP-10325-P-A Model Features

The Westinghouse LOCA M&E release model (Reference 6) was approved in February 1987. Westinghouse identified the need to clarify two model features: 1) the assumptions placed on the SG exit steam enthalpy during the post-reflood period, and 2) the assumed power level used in the LOCA M&E analysis. Resolution of this issue, i.e., NRC acceptance, is documented in Reference 7. The NRC staff found that the assumption used for the SG exit enthalpy during the post-reflood period is conservative. The NRC staff also found that Westinghouse's understanding of performing analyses at the licensed core power, regardless of power level, was an acceptable method, as long as the plant-specific calorimetric uncertainty is considered. The safety analysis code inputs for SQN were not directly affected.

5) Main Feedwater Addition Following a Reactor Trip

The Westinghouse methods account for the addition of main feedwater (MFW) to the SG following a LOCA in the time frame from reactor trip until MFW isolation is calculated to occur. Full MFW flow is assumed prior to the reactor trip. A review called into question the current modeling of the isolation of the SG secondary side on a reactor trip signal. The continued addition of MFW after reactor trip is adding energy to the secondary side above 212ºF and therefore in the long-term this additional energy would be released to the containment. This issue affected the current analysis and was corrected in the proposed analysis.

6) AFW System Purge and Unisolatable Volumes

After isolation of the MFW, a volume of hot MFW will reside in the main feed lines between the auxiliary feedwater (AFW) injection point and the SG secondary side. Once AFW flow is initiated, the hot MFW water would be pushed into the SG secondary side. As the SGs are calculated to depressurize, there may be additional volume of hot feedwater trapped between the AFW injection point and the MFW isolation valve that would flash and be pushed into the SG secondary. These two concerns were not considered in the LOCA M&E release models. This issue affected the current analysis and was corrected in the proposed analysis.

7) AFW Flow for FROTH Code

The LOCA M&E release analyses model AFW flow to the SGs because SG secondary conditions affect the LOCA M&E release magnitude and timing. In some of these analyses, the actual AFW flow used was total pump flow and not flow per SG. This resulted in SG AFW flows that were high. AFW flow is only modeled in the FROTH code which calculates the transient from end of reflood until the SG secondary side pressure has been calculated to have depressurized to the containment design pressure. This time range of between 200 and 1000 seconds varies depending upon plant type (i.e., 2, 3, or 4 loop), power level and containment design pressure. AFW was correctly modeled in the current analysis, thus SQN was not directly affected.

8) Possibility of Asymmetric AFW Flow

LOCA analyses are performed assuming that off-site power is lost coincident with the event, and with the limiting single failure of one diesel generator to start. If the plant design does not start the turbine-driven AFW pump on the loss-of-offsite power or a safety injection (SI) signal, the typical design would have one motor-driven AFW pump in operation which would not feed all SGs. Thus, one or more SGs may not receive any AFW flow. If the broken loop SG is the SG with no AFW flow, there will be an effect on the calculated LOCA M&E release. The current LOCA M&E release models do not contain a provision to model asymmetric AFW flow. Instead, this effect is bounded by the assumption of no AFW delivery. AFW was correctly modeled in the current analysis, thus SQN was not directly affected.

RAI-2

Reference 3 [NSAL-06-6], states:

> *"The issues were evaluated for ice condenser plants where the containment pressure is controlled by the melting of the ice inside containment. Instead of applying the impact in a pressure increase, the penalty was converted into an energy value. Benefits were found in the calculation of the SG secondary mass, and other analysis inputs, that were greater than the additional energy."*

Please explain how the impact on containment peak pressure is prevented and the penalty converted into energy.

TVA RESPONSE:

Relative to the NSAL issues and the resulting effect on ice condenser containment pressure analysis, sensitivity studies were performed to quantify the ice weight required to eliminate the pressure increase resulting from the NSAL-06-6 (Reference 3) issues. This additional ice weight penalty was converted into energy using heat of fusion calculations.

Conservatisms were found in the calculation of the SG secondary mass, decay heat, and other analysis inputs, that were greater than this additional energy, thus offsetting the effect on containment peak pressure.

All NSAL-06-6 issues were specifically addressed for SQN in the reanalysis.

RAI-3

Reference 1, Attachment 5 to its Enclosure, Table 1-2 provides the decay heat data in the proposed analysis. Reference 4, Enclosure 4, Table 1-2 lists the currently used decay heat data. Comparing the data used for the current and the proposed analysis, it is noted that for the first 10,000 seconds the decay heat is less in the proposed analysis than in the current analysis. Please note that the conservatism in the decay heat is important during the first 10,000 seconds because as per Reference 1, Attachment 5, Appendix D, the peak containment pressure of 11.33 psig occurs at about 6371 seconds.

(a) Please explain why the decay heat for the first 10,000 seconds is less in the proposed decay heat data than from the currently used data.

(b) Please list all the differences between the assumptions used for the decay heat calculation in the current and proposed analysis and justify if the assumptions used in the proposed analysis are less conservative.

TVA RESPONSE:

Response to Part a

In the time period between the current analysis (Reference 4) and the proposed analysis (Reference 1), the method of calculating the decay heat changed from a hand calculation to a calculation performed by a verified computer code. The two methods of calculating decay heat did not agree. Subsequent investigation through the Westinghouse corrective actions process showed that while the hand calculation of decay heat contained an error, the decay heat curves calculated by hand were conservative. The computer code calculation determined that decay heat is lower during the first 10,000 seconds than predicted by the hand calculation. Subsequent analyses use the decay heat calculation performed by computer code.

Response to Part b

The inputs and assumptions used to calculate the decay heat have not changed between the current analysis and the proposed analysis. Note that the decay heat calculation conforms to the American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 5.1-1979, "Decay Heat Power in Light Water Reactors," and the standard's assumptions and conservatisms.

RAI-4

Reference 1, Attachment 5 to Enclosure, Section 2.2 input assumption 5 states the accumulator nitrogen mass of 3479 lbs is included in the calculation. In the current analysis in Reference 4, Enclosure 4, Section 2.2 input assumption 5 states the accumulator mass of 3676 lbs is included in the calculation. A reduction in the nitrogen mass added to the containment may reduce the conservatism in the containment peak pressure analysis. In case the conservatism is reduced, please explain and justify the reason for the difference in the nitrogen mass added to the containment in the proposed and the current analysis.

TVA RESPONSE:

The accumulator nitrogen mass released to containment in the proposed analysis (Reference 1) is more aligned with the predicted containment conditions than the nitrogen mass released to containment in the current analysis (Reference 4). The total initial accumulator nitrogen mass available is the same in the current and proposed analyses. However, the mass of nitrogen released to containment is slightly lower in the proposed analysis than the current analysis because the current analysis assumed that all the accumulator nitrogen was released to containment. In the proposed analysis, some of the accumulator nitrogen is retained in the accumulator due to containment backpressure.

RAI-5

Reference 1, Attachment 5 to Enclosure, Section 2.2, referring to input assumption number 11, please explain the differences in the upper containment volume used in the current and the proposed analysis and how these volumes are related with their initial temperatures used in the analysis. In case the conservatism in the proposed analysis is reduced by using initial temperature of 80°F in the upper compartment, please justify the use of a lower temperature in the proposed analysis instead of using the current analysis initial temperature assumption of 85°F, which is the lower limit specified in TS Limiting Condition of Operation (LCO) 3.6.1.5.

TVA RESPONSE:

The containment initial conditions discussed in input assumption number 11 (Section 2.2) are the same in both the current analysis (Reference 4) and the proposed analysis (Reference 1). Specifically, the containment upper volume temperature used in both analyses is 80°F, which is the result of the adjustment of the minimum upper compartment temperature limit of 85°F specified in Technical Specification Limiting Condition of Operation (LCO) 3.6.1.5.

The perceived difference in the containment upper volume temperature between the current analysis and the proposed analysis is due to the additional text in the proposed analysis which is not describing a change to the analysis input but is meant to better describe the determination of compartment initial conditions.

The LOTIC code for ice condenser containment model does not have a separate containment volume for the ice condenser upper plenum, which contains a mass of cold air that needs to be accounted for in the analysis as it affects the containment pressure response. The upper containment volume and temperature in the current and proposed analyses were adjusted to account for this mass of cold air in the ice condenser upper plenum and thus maximize the upper compartment air mass and compression ratio. This adjustment results in a conservative calculated peak containment pressure response.

RAI-6

Reference 1, Attachment 5 to Enclosure, Table 2-1, under "Ice Condenser" item number 17, provide justification for reducing the thickness of containment wall panels and containment shell steel from 0.4625 ft used in the current analysis to 0.0625 ft used in the proposed analysis. Is this conductor exposed to the outside containment temperature?

TVA RESPONSE:

The text change to structural heat sink number 17 in Table 2-1 does not reflect a change in the analysis value, but instead reflects the correction of a typographical error in the current analysis (Reference 4) documentation. The correct containment wall panel and containment shell steel thickness is 0.0625 feet. This conductor is not exposed to the outside containment temperature.

RAI-7

Reference 1, Attachment 5 to Enclosure, Appendix A, under the heading "LOCA Mass and Energy Release Phase," please reconsider defining "Blowdown" which states:

> *"Blowdown - the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium state at containment design pressure."*

The containment peak pressure should be less than its design pressure during an accident. As per the above definition, the **containment pressure** has reached the containment **<u>design pressure</u>** at the end of blowdown period.

TVA RESPONSE:

With respect to the mass and energy release portion of the analysis, as described in WCAP-10325 (Reference 6), the SATAN-VI code considers the blowdown period to end at the time that the reactor coolant system reaches an equilibrium state at the containment design pressure. Sensitivity analyses have been performed, as discussed in WCAP-10325, that show that assuming a maximum containment backpressure in the mass and energy release portion of the analysis is conservative with respect to containment response.

It should be noted that this conservative assumption is a mechanistic way of defining the end of blowdown in SATAN-VI, and is not present in the containment integrity portion of the analysis.

RAI-8

Please provide a discussion regarding the impact on the NPSH analysis for the pumps that draw water from the containment sump during the recirculation mode of operation during the postulated accidents. Reference 1 does not provide any information on this analysis.

TVA RESPONSE:

The inputs used in the NPSH analysis remain bounding (i.e., conservative) for the relevant results of the proposed analysis (Reference 1) as discussed below:

- The NPSH analysis does not credit any pressurization of containment to demonstrate adequate NPSH so the decrease in peak containment pressure between the current and proposed analyses does not change the containment pressure used in the NPSH analysis.

- The NPSH analysis uses a sump water elevation head based on 1.4 million pounds of ice melt which is only a fraction of the total ice melt predicted by the current analysis. The larger amount of ice melt predicted in the proposed analysis and the correspondingly higher sump water elevation is not needed to show adequate NPSH.

- The NPSH analysis uses a sump water temperature input that is higher than the sump water temperature results in the proposed analysis so no change to the sump water temperature input to the NPSH analysis is needed.

There are no other results from the proposed analysis that could affect the inputs to the NPSH analysis. Because all the inputs to the NPSH analysis bound the relevant results of the proposed analysis, there is no effect to the NPSH analysis from the proposed analysis.

References:

1. Letter from TVA to NRC, "Application to Modify Ice Condenser Technical Specifications to Address Revisions in Westinghouse Mass and Energy Release Calculation (SQN-TS-12-04)," dated July 3, 2013 (ADAMS Accession Number ML13199A281)

2. NSAL-11-5, "Westinghouse LOCA Mass and Energy Release Calculation Issues," dated July 26, 2011 (ADAMS Accession Number ML13239A479)

3. NSAL-06-6, "LOCA Mass and Energy Release Analysis," dated June 6, 2006

4. Letter from TVA to NRC, "Sequoyah Nuclear Plant (SQN) -Units 1 and 2 -Technical Specification (TS) Change No. 01-04, Revised Ice Weight," dated September 12, 2001 (ADAMS Accession Number ML012640365)

5. WCAP-8264-P-A, Revision 1, "Topical Report Westinghouse Mass and Energy Release Data for Containment Design," dated August 31, 1975

6. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," dated May 1, 1983

7. Letter from Herbert N. Berkow (NRC) to Mr. James A. Gresham (Westinghouse): Acceptance of Clarifications of Topical Report WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version" (TAC No. MC7980), dated October 18, 2005 (ADAMS Accession Number ML052660242)