TVA

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

May 22, 2012

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, Unit 1
Facility Operating License No. DPR-77
Docket No. 50-327

Subject: **Response to Nuclear Regulatory Commission (NRC) Third Request for Additional Information Regarding the Heavy Load Lifts and UHS One Time Change in Support of Unit 2 SGRP (TAC NO. ME7225)**

Reference:

1. Letter from Tennessee Valley Authority (TVA) to NRC, "Application to Revise Sequoyah Nuclear Plant, Unit 1, Operating License and Technical Specification 3.7.5, 'Ultimate Heat Sink,' to Support Sequoyah Nuclear Plant, Unit 2, Steam Generator Replacement Project (TS-SQN-2011-05)," dated September 29, 2011
2. Electronic Mail from NRC to TVA, "Sequoyah, Unit 1 LAR - Heavy Load Lifts and UHS One Time Change in Support of Unit 2 SGRP (TAC NO. ME7225)," dated January 13, 2012
3. Letter from TVA to NRC, "Response to NRC Request for Additional Information Regarding the Heavy Load Lifts and UHS One Time Change in Support of Unit 2 SGRP (TAC NO. ME7225)," dated February 10, 2012
4. Electronic Mail from NRC to TVA, "Sequoyah, Unit 1 LAR - Heavy Load Lifts and UHS One Time Change in Support of Unit 2 SGRP (TAC NO. ME7225)," dated February 2, 2012
5. Letter from TVA to NRC, "Response to NRC Second Request for Additional Information Regarding the Heavy Load Lifts and UHS One Time Change in Support of Unit 2 SGRP (TAC NO. ME7225)," dated March 5, 2012

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6. Electronic Mail from NRC to TVA, "Sequoyah Request for Additional Information - Heavy Load Lifts (TAC NO. ME7225)," dated May 7, 2012

The purpose of this letter is to provide the TVA response to the request for additional information (RAI) received on May 7, 2012 (Reference 6). These questions were regarding a license amendment request provided to the NRC by letter dated September 29, 2011 (Reference 1).

The Reference 1 amendment request, proposed to add a one-time license condition to the Sequoyah Nuclear Plant (SQN), Unit 1, Operating License (OL) for the conduct of heavy load lifts associated with the SQN, Unit 2, Steam Generator Replacement Project (SGRP). The one-time license condition establishes special provisions and requirements for the safe operation of SQN, Unit 1, while large heavy load lifts are performed on SQN, Unit 2. In addition, a one-time change to SQN, Unit 1, Technical Specification (TS) 3.7.5, "Ultimate Heat Sink," is proposed to implement additional restrictions with respect to maximum average Essential Raw Cooling Water System supply header water temperature during large heavy load lifts performed to support the SQN, Unit 2, SGRP.

By electronic mail (email) dated January 13, 2012 (Reference 2), the NRC forwarded an RAI regarding the proposed changes establishing the special provisions and requirements for large heavy load lifts on SQN, Unit 2. TVA responded to this first RAI by letter dated February 10, 2012 (Reference 3). In an email dated February 2, 2012 (Reference 4), the NRC forwarded a second RAI regarding the proposed changes, and TVA responded to this second RAI by letter dated March 5, 2012 (Reference 5). Subsequently, by email dated May 7, 2012 (Reference 6), the NRC forwarded a third RAI.

The Enclosure to this letter provides TVA's response to the third RAI. This response was discussed with the NRC on May 14, 2012. Based on the response to Question 2 of the Enclosure to this letter, Table 9-7 of the original submittal was updated as indicated. No other changes have been made to the Reference 1 license amendment request or to the first or second RAI responses provided in References 3 and 5.

In accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosure to the Tennessee State Department of Environment and Conservation. There are no regulatory commitments included in this submittal. If you have any questions, please contact Clyde Mackaman at (423) 751-2834.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on the 22nd day of May 2012.

Respectfully,

J. W. Shea
Manager, Corporate Nuclear Licensing

Enclosure:
Response to NRC Third Request for Additional Information Regarding the Heavy Load Lifts in Support of Unit 2 SGRP

cc (Enclosure):

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

ENCLOSURE

**TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT
UNIT 2**

**RESPONSE TO NRC THIRD REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE HEAVY LOAD LIFTS IN SUPPORT OF UNIT 2 SGRP**

NRC Question 1

Page 31 of the submittal dated September 29, 2011 [Agency-wide Document Access and Management System (ADAMS) Accession No. ML11273A169] (submittal), states that, "the OLS [outside lift system] will not collapse or drop a load while loaded or unloaded during the SSE." What is the probability of a seismic event exceeding the SSE [safe shutdown earthquake] (or, alternatively, the strength of the OLS) while the OLS is loaded? If the risk from a seismic event could be significant compared to the random lift drop, please include this risk in your change in core damage frequency [CDF], and large early release frequency [LERF] estimates.

TVA Response

The Sequoyah Nuclear Plant Updated Final Safety Analysis Report (UFSAR), Section 2.5.2.4, defines the Safe Shutdown Earthquake (SSE) maximum ground acceleration of 0.18 g or 176.5 cm/sec^2. Using the following figure from the EPRI report NP-6995-D, "Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue," the frequency of a seismic event exceeding the SSE is 3.0E-04 per year.

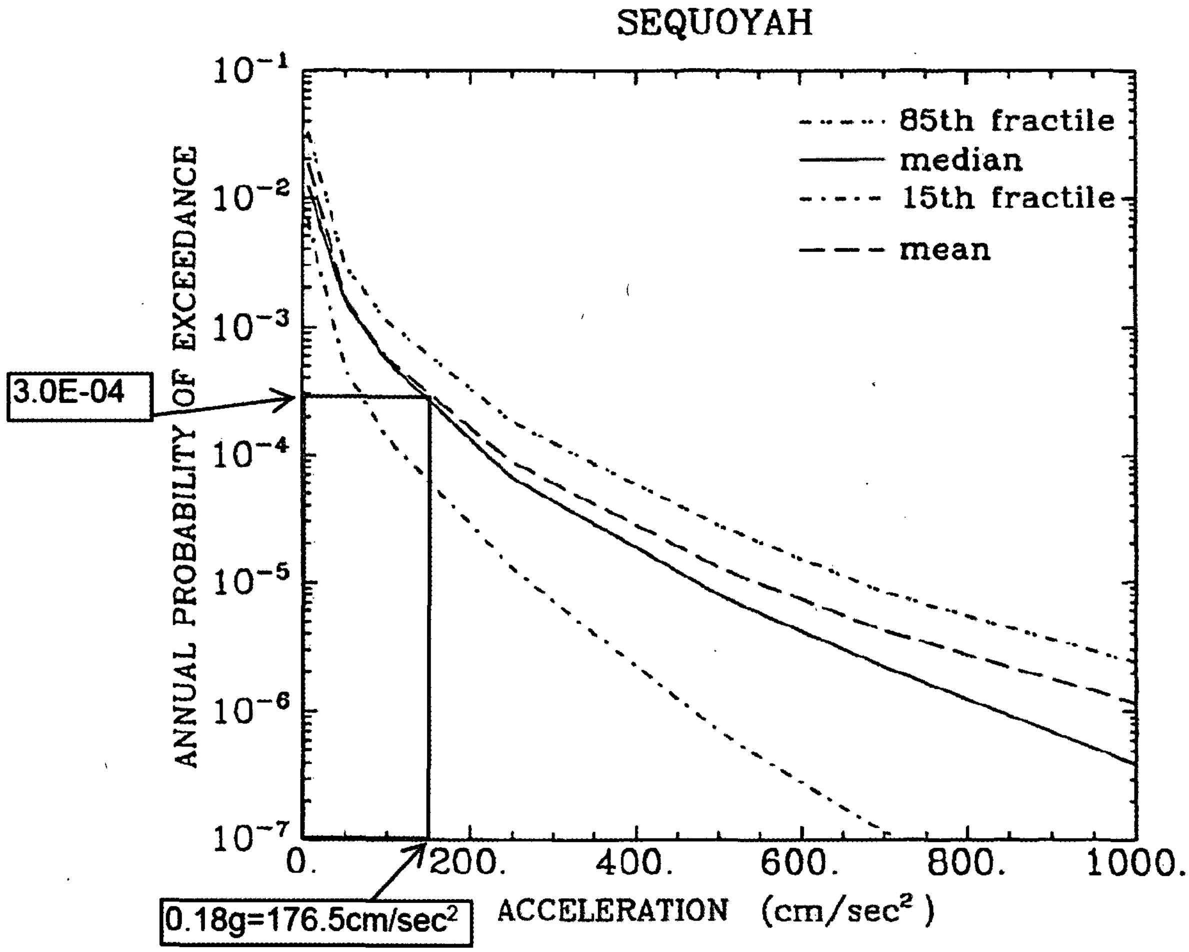

Figure 3-259. Annual probability of exceedance of peak ground acceleration: Sequoyah site.

Each load lift is estimated to take one hour to complete. Since there are eighteen load lifts, the total time during which the OLS will be loaded is eighteen hours. To calculate the probability of having a beyond SSE event while the OLS is loaded, the frequency of the seismic event exceeding the SSE (3.0E-4) was multiplied by the duration, in years, of the OLS being loaded (18 hrs / 8760 hrs per yr). The calculated probability of this event is 6.16E-07. This probability is well below the annual probability of having a random lift load drop, which is 1.017E-03, and is therefore not included in the Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) analysis.

NRC Question 2

Page 80 of the submittal states that an industry average *per-lift* *drop frequency of 5.6E-5 was used. Page 91 discusses upper and lower drop frequencies on a* *per-year* *basis. Page EA-1 of the supplement dated February 10, 2012 [ADAMS Accession No. ML12046A646] (RAI response) states that industry average load drop value of 5.6E-05* *per-year* *was used.*

a) Clearly define the values and the units of the lift drop failure parameter from the Boiling Water Reactor Owners Group (BWROG-TP-10-0XX) topical report that is used in your analysis.

b) If the value is a per-year value, address whether that value is applicable under these conditions (limited number of special lifts).

c) If the value is a per-lift [value], address why the 18 hour exposure time is used to calculate the Conditional Risk in Table 9-7 of the submittal.

TVA Response

a) The load drop frequency is calculated based on a per lift basis, the value in the BWROG topical report is 5.65E-05 per lift. With eighteen lifts being performed in the year, the frequency of having a load drop is 1.017E-03 per year.

b) The average load drop value in the BWROG topical report is per lift, not per year. The analysis is based on utilizing the load drop value on a per lift basis.

c) As discussed in the original submittal it is expected that each load lift will take one hour, and with eighteen lifts being performed, the expected total duration that a heavy load will be suspended is eighteen hours. Eighteen hours were used to be consistent with the evaluation of risk impact as discussed in Regulatory Guide 1.177, "An Approach For Plant-Specific, Risk Informed Decision Making: Technical Specifications," and to demonstrate that this one time technical specification change is less than the Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large Early Release Probability (ICLERP) thresholds of 1E-06 and 1E-07, which is considered small per the Regulatory Guide.

   Based on the initiating event frequency discussed earlier, the results presented in Table 9-7 of the original submittal were updated to:

Risk Metric	*SG Load Drop Value*	*Alignment Change Value*	*Exposure*	*Conditional Risk*
CDF	3.0643E-05	3.0591E-05	18 hours	1.0685E-10
LERF	4.4212E-06	4.4126E-06	18 hours	1.7671E-11

NRC Question 3

Provide the conditional core damage probability (CCDP) and the conditional large early release probability (CLERP) following a load drop with the ERCW realigned systems (e.g. used in the submittal estimates).

TVA Response

To calculate the conditional probability of the load drop initiating event with the Essential Raw Cooling Water (ERCW) system realigned, the load drop initiating event was set to a frequency of 1.0 and the other initiating events were set to a frequency of 0 in the SQN Revision 0 Computer Aided Fault Tree Analysis (CAFTA) model. The model was then quantified for CCDP and CLERP. A summary of those results are presented below.

Risk Metric	*Conditional Probability for the Load Drop Initiating Event*
CCDP	1.2463E-04
CLERP	9.0312E-06

Assumption 5 from Enclosure 2 of the original submittal states that the dam being constructed in the pipe tunnel has an assumed failure probability of 1.0. A sensitivity analysis was performed on this failure rate, with a dam failure rate of 0.1. The results of the sensitivity analysis are presented in the table below.

Risk Metric	*Sensitivity Conditional Probability with Dam installed*
CCDP	1.4938E-05
CLERP	1.3045E-06

NRC Question 4

The submittal discusses many flooding events. The licensee appears to have presented 2 types of flooding events, flooding due to the heavy load drop modeled in Figure 9-7, and flooding caused by random failure unrelated to the heavy load movement represented by the 60 odd flooding initiating events in Tables 9-2 and 9-3.

Assuming random internal floods are modeled, summarize the initiating event frequencies in Tables 9-2 and 9-3, how the frequencies are developed, and whether all the floods are the same size. If random internal flooding events are not modeled, please explain what the initiating events in Tables 9-2 and 9-3 are, and how they are used in the risk analysis.

TVA Response

The table below provides the initiating event frequency for those initiating events presented in Tables 9-2 and 9-3 of Enclosure 2 of the original submittal (Reference 1 of the cover letter).

Initiating Event	*Frequency*
%653.0-A01_067_F_2A	2.678E-04
%653.0-A12_067_F_2A	1.848E-05
%653.0-A14_067_F_2A	2.260E-05
%669.0-A01_067_F_2A	5.256E-06
%669.0-A01_067_M_2A	8.337E-07
%669.0-A24_067_F_2A	1.761E-05
%669.0-A24_067_M_2A	2.961E-06
%669.0-A25_067_F_2A	2.800E-04
%669.0-A25_067_M_2A	4.430E-05
%669.0-A26_067_F_2A	1.730E-05
%690.0-A01-1_067_F_2A	8.770E-04
%690.0-A01-1_067_M_2A	1.390E-04
%690.0-A01-2_067_F_2A	1.120E-05
%690.0-A01-2_067_M_2A	1.770E-06
%690.0-A01-4_067_F_2A	1.566E-05
%690.0-A01-4_067_M_2A	1.390E-04
%690.0-A19_067_F_2A	3.000E-04
%690.0-A19_067_M_2A	4.750E-05
%690.0-A29_067_F_2A	6.690E-05
%690.0-A29_067_M_2A	1.660E-06
%714.0-A01-1_067_F_2A	1.027E-05
%714.0-A01-1_067_M_2A	1.970E-04
%714.0-A01-2_067_F_2A	3.653E-06
%714.0-A01-2_067_M_2A	1.970E-04
%714.0-A09_067_F_2A	1.160E-04
%714.0-A09_067_M_2A	1.830E-05
%734.0-A13-1_067_F_2A	1.240E-05
%734.0-A13-1_067_M_2A	1.970E-06
%734.0-A13-2_067_F_2A	1.240E-05
%734.0-A13-2_067_M_2A	1.970E-06
%653.0-A01_067_F_2B	2.678E-04
%653.0-A13_067_F_2B	1.310E-05

Initiating Event	*Frequency*
%653.0-A15_067_F_2B	1.440E-05
%669.0-A01_067_F_2B	5.256E-06
%669.0-A01_067_M_2B	8.337E-07
%669.0-A24_067_F_2B	1.761E-05
%669.0-A24_067_M_2B	2.961E-06
%669.0-A25_067_F_2B	2.800E-04
%669.0-A25_067_M_2B	4.430E-05
%669.0-A26_067_F_2B	2.240E-05
%669.0-A26_067_M_2B	3.550E-06
%690.0-A01-1_067_F_2B	8.770E-04
%690.0-A01-1_067_M_2B	1.390E-04
%690.0-A01-2_067_F_2B	1.110E-03
%690.0-A01-2_067_M_2B	9.040E-06
%690.0-A01-3_067_F_2B	5.980E-05
%690.0-A01-3_067_M_2B	9.040E-06
%690.0-A01-4_067_F_2B	1.158E-04
%690.0-A01-4_067_M_2B	1.390E-04
%690.0-A19_067_F_2B	3.000E-04
%690.0-A19_067_M_2B	4.750E-05
%690.0-A29_067_F_2B	6.690E-05
%690.0-A29_067_M_2B	1.660E-06
%714.0-A01-1_067_F_2B	1.027E-05
%714.0-A01-1_067_M_2B	1.970E-04
%714.0-A01-2_067_F_2B	1.040E-03
%714.0-A01-2_067_M_2B	1.970E-04
%714.0-A09_067_F_2B	1.160E-04
%714.0-A09_067_M_2B	1.830E-05
%734.0-A13-1_067_F_2B	1.240E-05
%734.0-A13-1_067_M_2B	1.970E-06
%734.0-A13-2_067_F_2B	1.600E-05
%734.0-A13-2_067_M_2B	2.550E-06

These initiating event frequencies were calculated for the Internal Flooding analysis for SQN and as such are not new internal flooding scenarios. For each flood area defined in the analysis, the linear feet of piping within this area was collected, and this value was multiplied by the appropriate initiating event frequency as described in the EPRI Report on pipe rupture frequencies (EPRI Report 1021086, Revision 2).

The flood sizes are based on the flow rates specified in the naming convention for the initiating event scenario descriptions. For a flood with the "_F_" in the name, the initiating event is a break where the flow rate out of the break is greater than 100 gallons per minute but less than 2,000 gallons per minute. For a flood event with a "_M_" in the name, the initiating event is break where the flow rate is greater than 2,000 gallons per minute up to the maximum flow rate out of the break. Therefore, each of the initiating events represents a wide range of potential pipe break sizes, but the analysis in the Probabilistic Risk Assessment (PRA) uses the most limiting flow rates (i.e., 2,000 gallons per minute for floods "_F_", and the maximum flow rates for major floods "_M_").

NRC Question 5

Assumption 6 in section 9.3 in the submittal states that:

> *[t]here are no operator actions which could isolate the flooding event caused by the load drop prior to impacting the RHR [residual heat removal] and CS [containment spray] pumps on elevation 653.*

Address whether any operator actions are credited to isolate broken piping after a load drop. If operator actions are credited, provide a summary for each.

TVA Response

For the purpose of the analysis, operator actions are credited for being able to stop the flooding event. Assumption 6 was included to show that any operator actions to stop the flooding event will not be successful in preventing damage to the RHR and CS pumps. During a load drop event, the volume of water within the RWST is significant enough to cause the RHR and CS pumps to be unavailable for accident mitigation due to submergence.

After the load drop, operators will be required to mitigate the flooding event by using the steps outlined in Abnormal Operating Procedure AOP-M.01, "Loss of Essential Raw Cooling Water." Specifically, they will have to isolate the "B" discharge header of the ERCW system that is an action modeled within the PRA. This action would need to be performed prior to the flood waters impacting the 669 foot elevation of the Auxiliary Building. The volume of water required to completely fill the passive sump and the rooms on the 653 foot elevation would be greater than 1.5 million gallons. After a load drop severing the piping, it will take approximately 75 minutes for the water level to reach the 669 foot elevation. There will be ample time to perform operator actions to mitigate the flood prior to impacting the equipment on the 669 foot elevation, and all necessary isolation valves would be accessible.