TVA

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

February 10, 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 NRC 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 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

By letter dated September 29, 2011 (Reference 1), the Tennessee Valley Authority (TVA) submitted a one-time request for amendment to the Operating License (OL) for Sequoyah Nuclear Plant (SQN), Unit 1. The amendment request proposed to add a one-time license condition to the SQN, Unit 1, OL for the conduct of heavy load lifts associated with the SQN, Unit 2, Steam Generator Replacement Project (SGRP).

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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 dated January 13, 2012, (Reference 2), the NRC forwarded a request for additional information (RAI) regarding the proposed changes. As agreed, the response to the RAI is due 30 days from January 17, 2012 since the RAI was issued after normal working hours on Friday, January 13, 2012, with the next normal work day being Tuesday, January 17, 2012, due to the intervening weekend and holiday. The enclosure to this letter provides the TVA response to the NRC RAI. No changes have been made to the enclosures submitted with the request for amendment to the OL and TSs by Reference 1.

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 10th day of February 2012.

Respectfully,

J. W. Shea
Manager, Corporate Nuclear Licensing

Enclosure:

Response to Request for Additional Information Regarding the Heavy Load Lifts and UHS One Time Change in Support of Unit 2 SGRP

cc (Enclosures):

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 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING THE HEAVY LOAD LIFTS AND UHS ONE TIME CHANGE IN SUPPORT OF UNIT 2 SGRP

NRC Question

In the submittal, the licensee has identified the change in risk associated with the requested ERCW alignments as 7.2E-7/year (CDF) and 4.05E-8/year (LERF); this is the risk increase above the nominal risk of maintaining the normal ERCW alignment. Also, the licensee has identified the additional risk of performing heavy load lifts as 7.0E-9/year (CDF) and 5.0E-10/year (LERF), given that the ERCW is in its proposed special alignment. The risk increase incurred due solely to the special ERCW alignment is two orders of magnitude greater than the heavy load lift risk increase. It is assumed by the NRC staff that if the ERCW is not aligned as described, then the risk increase associated with the heavy load lifts would be significant, or at least greater than the 7.2E-7/year (CDF) and 4.05E-8/year (LERF) reported for the ERCW alignment. However, the risk metrics associated with performing the same heavy load lifts with ERCW in its normal alignment was not provided in the submittal, which would permit the NRC staff to assess the relative risk impacts. Therefore, the NRC staff cannot conclude that the special ERCW alignment is necessary to avoid a more significant risk increase during the heavy load lifts.

The licensee is requested to justify the need for the ERCW alignment and the resulting risk increases incurred by providing the risk metrics associated with heavy load lifts if the normal ERCW alignment is maintained.

TVA Response

The need for the special ERCW alignment is required in order to meet the requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Section 5.1.5 of NUREG-0612 provides evaluation guidelines which include analysis of the effects of load drops (in accordance with Appendix A to NUREG-0612) and the results should indicate that damage to safe shutdown equipment is not sufficient to preclude safe shutdown. There are three affected sections of Essential Raw Cooling Water (ERCW) system piping which are postulated to fail from a load drop associated with the conduct of the SQN, Unit 2, heavy load lifts for the Steam Generator Replacement Project (SGRP). These ERCW piping sections are the "2A" and "2B" ERCW supply headers and the "B" ERCW discharge header. These piping sections are located in the pipe tunnel that runs through the drop zone shown in Figure 6-2 of Enclosure 2 to Reference 1. In the event of a large heavy load drop, the worst case damage to the ERCW system piping running through the pipe tunnel is assumed to be severing each of the "2A" and "2B" ERCW supply headers and crushing the "B" ERCW discharge header piping such that no flow is possible through the discharge header. In order to ensure that adequate heat removal capability is maintained if this event were to occur, the ERCW system is required to be pre-aligned before commencement of large heavy load handling that could potentially damage this piping. Details of this discussion are provided in Section 8.3, "Heavy Load Drop Protection

Plans/Compensatory Measures, Essential Raw Cooling Water System," of Enclosure 2 to the letter from TVA to the NRC dated September 29, 2011 (Reference 1 of cover letter).

The risk metrics associated with SGRP heavy load lifts with ERCW in its normal alignment are provided in the Attachment to this Enclosure. Performing the steam generator load lift while in the normal plant configuration will result in a Core Damage Frequency (CDF) increase of greater than 1E-5/year, which is Region I of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Risk increases that fall within Region I are not allowed; consequently, TVA developed the alternative ERCW heavy load lift alignment proposed in the license amendment request.

ATTACHMENT

PRA EVALUATION

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING THE HEAVY LOAD LIFTS AND UHS ONE TIME CHANGE IN SUPPORT OF UNIT 2 SGRP

1.0 Technical Adequacy of the Probabilistic Risk Assessment (PRA) Model

The current PRA model for SQN was issued on May 27, 2011. The previous model was updated from RISKMAN to CAFTA, which included a detailed internal flooding analysis, and a more in depth analysis of all plant systems to represent the as built, as operated plant. A peer review of the model was performed in January 2011 on the internal events and internal flooding supporting requirements of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard. The peer review team endorsed the model as being compliant with the ASME/ANS PRA standard with 77 facts and observations identified. All of the Facts and Observations were resolved to meet capability category II prior to issue of the model of record (Reference 4).

For the SGRP, all the postulated accidents and system alignments are covered in the PRA model. The PRA model is based on the as built, as operated plant design and can be used to evaluate a possible load drop event by applying specific modeling changes. A detailed analysis of where a load drop can occur and what systems can be affected is documented in the SQN Unit 2 Steam Generator Replacement Rigging and Heavy Load Handling Technical Report (Reference 5).

2.0 References

1. BWROG-TP-10-0XX, "BWROG Integrated Risk-Informed Regulation Committee Guidance Document, Configuration Risk Management - Heavy Loads," Revision C, April 2010.
2. MDQ000-067-2000-0095, "Evaluation of ERCW System Effects from a Unit 2 Steam Generator Replacement Heavy Load Drop," Revision 11.
3. MDN-000-000-2010-0203, "SQN Probabilistic Risk Assessment – Internal Flooding Analysis."
4. MDN-000-000-2010-0200, "SQN Probabilistic Risk Assessment – Summary Notebook."
5. SQN2-SGR-TR1, "SQN Unit 2 Steam Generator Replacement Rigging and Heavy Load Handling Technical Report," Revision 2.

3.0 Assumptions

1. The frequency used for the load drop initiator is the industry average value of 5.6E-05/year from the Boiling Water Reactor Owners Group (BWROG) Integrated Risk-Informed Regulation Committee Guidance Document (Reference 1).

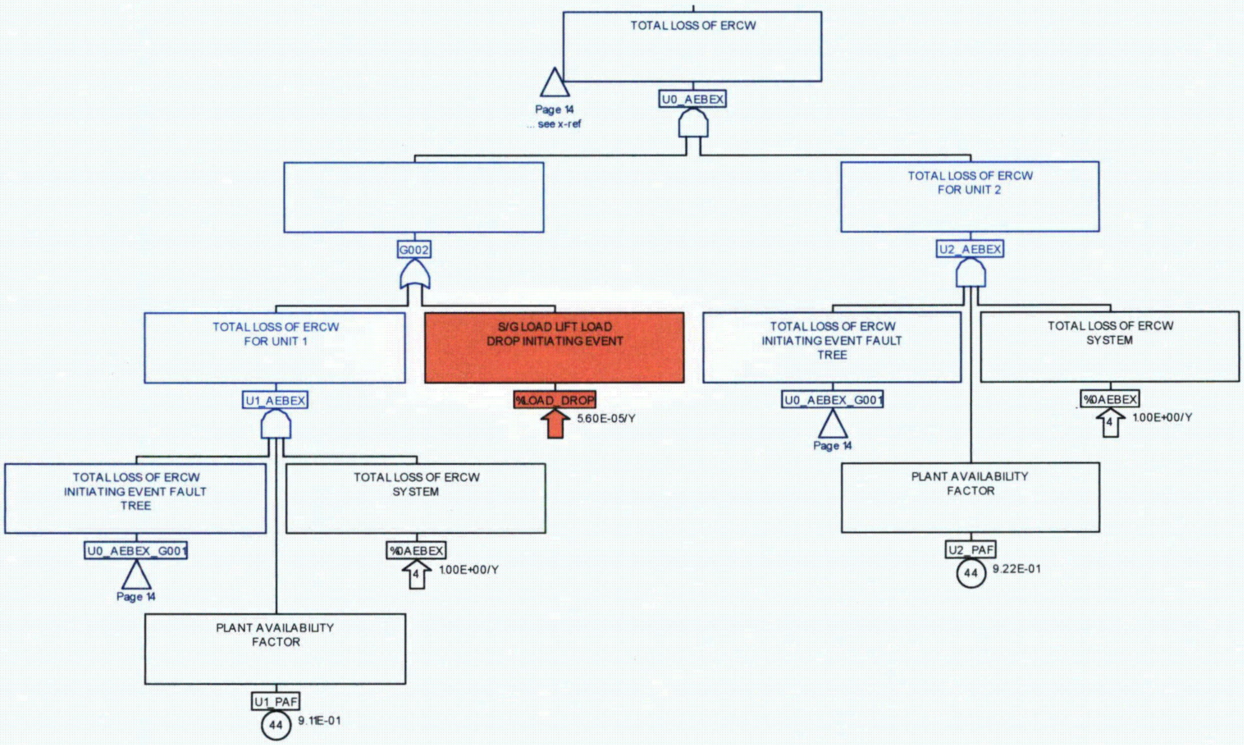
2. Due to the ERCW crosstie between SQN, Units 1 and 2, the load drop is assumed to depressurize the entire ERCW system. This event is equivalent to a total loss of ERCW initiating event, and it is assumed that there are no operator actions which can mitigate the loss of ERCW to the 1A and 1B ERCW supply headers (Reference 2).

3. There are no actions that can be taken to isolate the flood prior to damaging the Residual Heat Removal (RHR) and Containment Spray pumps on elevation 653 feet (Reference 3).

4.0 Probabilistic Risk Assessment Modeling

For this evaluation a new initiating event was added to the model (%LOAD_DROP). This event is equivalent to the total loss of ERCW initiating event and as such was modeled as an OR gate with the logic for the Unit 1 total loss of ERCW initiating event, see Figure 1.

Figure 1 – Model Change for Load Drop



5.0 Results

The change in CDF and Large Early Release Frequency (LERF) are presented in Table 1, and the Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large Early Release Probability (ICLERP) are presented in Table 2.

Table 1 – Conditional Risk Increase for S/G Load Drop

Risk Metric	*Model of Record Value*	*Load Drop Value*	*Change in Value*
CDF	3.0231E-05	4.1513E-05	1.1282E-05
LERF	4.3899E-06	4.7114E-06	3.2150E-07

Table 2 – Incremental Risk Increase for S/G Load Drop

Risk Metric	*Load Drop Value*	*Model of Record Value*	*Exposure*	*Conditional Risk*
CDF	4.1513E-05	3.0231E-05	18 Hours	2.3225E-08
LERF	4.7114E-06	4.3899E-06	18 Hours	6.6062E-10

As shown in the results table, a load drop while in the normal ERCW configuration would have a large impact on Unit 1. While the Auxiliary Feedwater systems are available initially, without seal cooling via the ERCW/Component Cooling System (CCS) heat exchangers, a resultant seal Loss of Coolant Accident (LOCA) will occur. The RHR recirculation mode of cooling is impossible due to the submergence of the RHR pumps from internal flooding and loss of cooling to the CCS heat exchangers, resulting in core damage.

The CDF results as presented above fall within Region I of the Acceptance Guidelines in Regulatory Guide 1.174. Any applications which cause a CDF increase of greater than 1E-5/year are not normally considered.

6.0 Uncertainties

6.1 Evaluation of Assumptions in Section 3.0

6.1.1 Assumption 1

The frequency of the load drop used in the analysis comes from an industry average per lift drop estimate. Since the change in initiating event frequency is between the upper and lower bounds mentioned in Reference 1 and the overall impact of the load drop is large, this assumption has low impact on the results and no sensitivity analysis was performed.

6.1.2 Assumption 2

This is a conservative assumption based on the present plant procedures and timing developed for Human Action Basic Event FLAB67CM. Due to insufficient timing available prior to submergence of the ERCW and Containment Spray pumps, the restoration of the ERCW headers will have minimal impact on the analysis.

6.1.3 Assumption 3

This is a conservative assumption that is based on the worst case scenario load drop within the pipe tunnel. By failing both the Refueling Water Storage Tank (RWST) and ERCW lines the volume of water entering the Auxiliary Building would be such that all components at the base elevation 653 feet would fail.