TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

April 14, 2008

TVA-SQN-TS-08-01 10 CFR 50.90

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

Gentlemen:

In the Matter of	)	Docket No. 50-327
Tennessee Valley Authority (TVA)	)	

SEQUOYAH NUCLEAR PLANT (SQN) - UNIT 1 - TECHNICAL SPECIFICATION (TS) CHANGE - 08-01 "REVISION OF CORE OPERATING LIMITS REPORT (COLR) REFERENCES FOR REALISTIC LARGE BREAK LOSS OF COOLANT ACCIDENT METHODOLOGY"

Pursuant to 10 CFR 50.90, TVA is submitting a request for a TS change (TSC-08-01) to License DPR-77 for SQN. The proposed TS change will add a new reference in TS Section 6.9.1.14.a. The new reference is "EMF-2103P-A, 'Realistic Large Break LOCA Methodology for Pressurized Water Reactors.'" This change is similar to the Unit 2 change requested in TSC-07-04 dated July 26, 2007. The changes submitted to support the Unit 2 realistic large break LOCA methodology review have been incorporated into the Unit 1 analysis.

Enclosure 1 is a description and justification of the proposed amendment. Annotated versions of the affected TS pages are provided in the attachment. Enclosure 2 provides the plant specific analysis for the application of the revised methodology to SQN. Portions of Enclosure 2 are proprietary to AREVA Nuclear Power (NP). Enclosure 3 provides a non-proprietary version of the document contained in Enclosure 2.

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Accordingly, Enclosure 4 includes the AREVA NP Application for Withholding Proprietary Information from Public Disclosure, and an accompanying Affidavit signed by AREVA NP, the owner of the information. Also included are a Proprietary Information Notice and a Copyright Notice. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission, and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. TVA respectfully requests that the AREVA NP proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS change qualifies for 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 enclosures to the Tennessee State Department of Public Health.

The proposed change is necessary for the planned core design for the Unit 1 Cycle 17 operation in the spring of 2009. TVA held discussions with NRC and determined that the proposed schedule for review and approval was reasonable and achievable. Therefore, TVA requests approval of this TS change by March 2009 to support the Unit 1 refueling outage and that the implementation of the revised TS be within 60 days of NRC approval.

There are no regulatory commitments associated with this submittal. If you have any questions about this change, please contact me at (423) 843-6672.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 14th day of April, 2008.

Sincerely,

Russell R. Thompson

Russell R. Thompson
Site Licensing Supervisor

Enclosures:
1. Evaluation of the Proposed Change
2. Proprietary Version of SQN's Plant Specific Topical
3. Non-Proprietary Proprietary Version of SQN's Plant Specific Topical
4. AREVA NP Affidavit for Withholding of Proprietary Information

cc: See page 3

Enclosures
cc (Enclosures):

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ENCLOSURE 1

EVALUATION OF THE PROPOSED CHANGE
SEQUOYAH NUCLEAR PLANT (SQN) UNIT 1
REVISION OF CORE OPERATING LIMITS REPORT (COLR) REFERENCES FOR REALISTIC LARGE BREAK LOSS OF COOLANT ACCIDENT (LOCA) METHODOLOGY

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License DPR-77 for SQN Unit 1.

The proposed changes would revise the Operating License(s) to incorporate into the Core Operating Limits Report (COLR) a reference for realistic large break (LB) LOCA methodology. The proposed change is necessary for the planned core design for the Unit 1 Cycle 17 operation in the spring of 2009.

2.0 DETAILED DESCRIPTION

This letter is a request to amend Operating License DPR-77 for SQN Unit 1. The proposed change will revise the list of topical reports used to prepare the core operating limits report by adding a new methodology for LB LOCA that utilizes a realistic analysis methodology. The proposed changes will add a new reference in TS Section 6.9.1.14.a. The new reference is "EMF-2103P-A, 'Realistic Large Break LOCA Methodology for Pressurized Water Reactors." This change is requested to support core loading designs for Unit 1 fuel-load configurations in future operating cycles.

3.0 TECHNICAL EVALUATION

The NRC safety evaluation report (SER) for the realistic LB LOCA methodology, EMF-2103P-A, states, "The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses, including the calculated worst break size, PCT, and local and total oxidation."

AREVA NP has performed a plant specific realistic LB LOCA analysis for SQN Unit 1 using the NRC approved methodology in EMF-2103P-A. An explanation of the analysis and results are presented in the Enclosure 2 (proprietary) and Enclosure 3 (non-proprietary) reports (ANP-2695).

The information in the report is similar in scope and format to information provided for previous AREVA realistic LB LOCA plant specific applications (i.e., H. B. Robinson, Fort Calhoun, and Palisades). The changes submitted to support the Unit 2 realistic LB LOCA methodology have been incorporated into the Unit 1 analysis. Section 3.1 of the report describes the postulated LB LOCA event. Section 3.2 describes the models used in the analysis. The plant general arrangement and system parameters used in the analysis are described in Section 3.3. Compliance with the generic methodology SER is described in Section 3.4. Section 3.5 summarizes the results of the analysis.

The analysis assumes full core power operation at 3479 MWt (current rated thermal power with maximum measurement uncertainty applied), a uniform steam generator tube plugging level of 15 percent, a total core peaking factor (FQ) of 2.65 (including uncertainties), and a nuclear enthalpy rise factor (FΔh) of 1.706 (including uncertainty). The analysis also addresses typical operational ranges for pressurizer pressure and level; accumulator pressure, temperature and level; core average temperature; core flow; containment temperature and pressure; and refueling water storage tank temperature. The realistic LB LOCA results are based on a case set of 59 individual transient cases. The results demonstrate the adequacy of the emergency core cooling system (ECCS) to meet the performance acceptance criteria established by 10 CFR 50.46(b). The limiting calculated fuel peak clad temperature established by the analysis is 1,809 degrees Fahrenheit.

One of the limitations specified in the NRC SER states, "The model does not determine whether Criterion 5 of 10 CFR 50.46, long term cooling has been satisfied. This will be determined by each applicant or licensee as part of its application of this methodology." For SQN, the long-term cooling analysis is acceptable and not affected by this submittal. The current post-LOCA long-term reactor core cooling analysis was performed by Westinghouse in 2001 to address refueling water storage tank (RWST) and cold leg accumulator boron concentration changes associated with the tritium production core. The results of the analysis were summarized in Section 2.15.5.5 of AREVA Topical Report No. BAW-10237, that was submitted to NRC for review as part of the Sequoyah tritium production license amendment request (i.e., SQN TS Change Request No. TVA-SQN-TS-00-06) dated September 21, 2001.

The post-LOCA long-term cooling analysis involves calculations that ensure boron precipitation does not occur in the reactor vessel (also referred to as the hot leg switchover analysis) and confirms the post-LOCA ECCS performance in both the hot leg and cold leg recirculation mode is sufficient to prevent core heatup. The hot leg switchover analysis establishes the hot leg switchover time based on an established boron precipitation limit for the sump recirculation inventory. This analysis is governed by the limiting volume and boron concentration for the various sources of water which contribute to the post-LOCA sump recirculation inventory (i.e., accumulators, RWST, ice condenser melt and reactor coolant system volume). The analysis is typically only reanalyzed when one of these parameters (volume or boron concentration) changes. The ECCS performance analysis confirms that there is sufficient ECCS flow to exceed the core boil-off rate based on a conservative core decay heat assumption at the time of ECCS switchover from injection mode to sump recirculation mode. This analysis is governed by decay heat, ECCS minimum pump performance requirements, and pump alignment assumptions. The analysis is typically only reanalyzed when one of these characteristics change.

For the SQN plant-specific application of the AREVA realistic LB LOCA analysis, there are no changes to (1) the rated core power affecting post-LOCA decay heat, (2) the limiting volumes or boron concentrations for the constituent parts of the post-LOCA sump recirculation inventory, and (3) ECCS system performance or operational alignments. As such, the existing long-term core cooling analysis remains conservative and bounding for the conditions analyzed by the SQN realistic LB LOCA analysis.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

This letter is a request to amend Operating License DPR-77 for SQN Unit 1. The proposed changes will add a new reference in TS Section 6.9.1.14.a. The new reference is "EMF-2103P-A, 'Realistic Large Break LOCA Methodology for Pressurized Water Reactors.'"

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission's regulatory requirements related to the content of the TS are contained in Title 10, Code of Federal Regulations (10 CFR), Section 50.36. The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings, and control settings; (2) limiting conditions for operation (LCO); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The requirements for the initiation of a reactor trip resulting from a turbine trip are included in the TS in accordance with 10 CFR 50.36(c)(2), "Limiting Conditions for Operation."

As stated in 10 CFR 50.59(c)(1)(i), a licensee is required to submit a license amendment pursuant to 10 CFR 50.90 if a change to the technical specification (TS) is required. Furthermore, the requirements of 10CFR 50.59 necessitate that the NRC approve the TS changes before the changes are implemented. TVA's submittal meets the requirements of 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90.

Section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," specifies requirements for the acceptability of the ECCS. Paragraphs 50.46(a)(1)(i) and 50.46(a)(1)(ii) of 10 CFR specify alternative approaches to show compliance with the acceptance criteria of 10 CFR 50.46(b). Part 50 of 10 CFR, Appendix K, provides requirements for calculating whether those acceptance criteria are satisfied. Compliance with these criteria demonstrates the acceptability, following a LOCA, of (1) the peak calculated cladding temperature, (2) the maximum cladding oxidation, (3) the maximum hydrogen generation, (4) the capability to maintain a coolable geometry, and (5) the capability to maintain long-term core cooling. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," dated May 1989, provides guidance on methods acceptable to the NRC staff for realistic or best-estimate calculations of ECCS performance during a LOCA. Technical Branch Position CSB 6-1, "Minimum Containment Pressure Model for PWR [Pressurized-Water Reactor] ECCS Performance Evaluation," of NUREG-0800, the Standard Review Plan, provides guidance for complying with Appendix K, Section I.D.2. These regulatory documents provide the overall requirements and recommendations for ECCS modeling and acceptable methodologies to ensure the capability to mitigate the consequences of postulated events. The proposed change is consistent with the requirements and guidance of these documents and only modifies the methodology used to evaluate the LB LOCA event. The proposed use of the AREVA NP realistic methodology for LB LOCAs continues to meet the requirements of the applicable regulatory documents and will not result in an adverse impact to nuclear safety.

4.2 Precedent

The proposed SQN change is consistent with AREVA NP's NRC approved Topical EMF-2103P-A, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," as evaluated in AREVA NP Topical ANP-2695P, "Sequoyah Unit 1 Nuclear Plant Realistic Large Break LOCA Analysis." Similar changes have been previously requested and approved by NRC for SQN Unit 2 on April 4, 2008, H. B. Robinson Steam Electric Plant in September 2006, and Fort Calhoun Station in November 2006. Palisades Nuclear Plant's submittal is currently under review by NRC.

4.3 Significant Hazards Consideration

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change adds an approved analytical method for evaluating a large break (LB) loss of coolant accident (LOCA). The proposed change will not affect previously evaluated accidents because they continue to be analyzed by NRC approved methodologies to ensure required safety limits are maintained. The acceptance criteria of the SQN Final Safety Analysis Report analyzed accidents and anticipated operational occurrences are not affected by the proposed addition of the realistic LB LOCA methodology. As the evaluations for accidents and operation occurrences are not adversely affected, the proposed change will not increase the consequences of a postulated event.

   The proposed change does not result in any modification of the plant equipment or operating practices and therefore, does not alter plant conditions or plant response prior to or after postulated events.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   As previously noted, the proposed change does not result in any modification of the plant equipment or operating practices and therefore, does not alter plant conditions or plant response prior to or after postulated events.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change does not alter plant equipment including the automatic accident mitigation setpoints designed to mitigate the affects of a postulated accident. The accident analyses and plant safety limits continue to be acceptable as evaluated by NRC approved methodologies. The proposed application of the realistic LB LOCA methodology ensures acceptable margins and limits for fuel core designs.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 <u>Conclusions</u>

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or SR. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. EMF-2103P-A, Revision 0, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” dated April 2003.
2. ANP-2655P, Revision 0, “Sequoyah Unit 2 Nuclear Plant Realistic Large Break LOCA Analysis,” dated June 2007.
3. ANP-2655P, Revision 1, “Sequoyah Unit 2 Nuclear Plant Realistic Large Break LOCA Analysis,” dated February 2008.

ATTACHMENT

Technical Specifications Page Markup

ATTACHMENT

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

I. AFFECTED PAGE LIST

Unit 1

6-13a

II. MARKED PAGES

See attached.

Insert

9. *EMF-2103P-A, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors"*

Insert

CORE OPERATING LIMITS REPORT (continued)

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code"

6. WCAP-10266-P-A, "The 1981 Revision of Westinghouse Evaluation Model Using BASH CODE"

7. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel"

8. BAW-10186-A, "Extended Burnup Evaluation"

6.9.1.14.b The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle or within 30 days of issuance of any midcycle revision of the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS (PTLR) REPORT

6.9.1.15 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

Specification 3.4.9.1, "RCS Pressure and Temperature (P/T) Limits"

Specification 3.4.12, "Low Temperature Over Pressure Protection (LTOP) System"

6.9.1.15.a The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. Westinghouse Topical Report WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

2. Westinghouse Topical Report WCAP-15293, "Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation."

3. Westinghouse Topical Report WCAP-15984, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2."

6.9.1.15.b The PTLR shall be provided to the NRC within 30 days of issuance of any revision or supplement thereto.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.16 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 6.8.4.k, Steam Generator (SG) Program. The report shall include: