February 7, 2002

Mr. J. A. Scalice
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT (SQN), UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION ON TECHNICAL SPECIFICATION CHANGE NO. 01-08, "INCREASE MAXIMUM ALLOWED REACTOR POWER LEVEL TO 3455 MEGA-WATT THERMAL (MWt)" (TAC NOS. MB3435 AND MB3436)

Dear Mr. Scalice:

The subject Technical Specification Amendment Request was submitted to the U.S. Nuclear Regulatory Commission (NRC) for review and approval on November 15, 2001, by the Tennessee Valley Authority (TVA). The proposed license amendment would increase the full core thermal power rating by 1.3 percent from 3411 MWt to 3455 MWt, based on planned installation of the improved Caldon, Incorporated Leading Edge Flow Meter feedwater flow measurement instrumentation. The NRC staff is in the process of reviewing TVA's submittal.

As discussed during a conference call on February 7, 2001, the NRC staff requires responses to the enclosed Request for Additional Information to proceed with its review. Following the call, Mr. Kieth Weller of the SQN Licensing Staff stated that TVA would respond to this request by March 11, 2001.

Please have your staff contact me at (301) 415-2010 if there are any questions regarding the enclosed request.

Sincerely,

/RA/

Ronald W. Hernan, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosure: Request for Additional Information

cc w/enclosure: See next page

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REQUEST FOR ADDITIONAL INFORMATION

1.3 PERCENT RATED POWER INCREASE

SEQUOYAH NUCLEAR PLANT

DOCKET NOS. 50-327 and 50-328

Request for Additional Information for the Review of Sequoyah Power Uprate Applications

1. Provide results of an Anticipated Transient Without Scram (ATWS) analysis demonstrating that the plant at power uprate conditions is within the bounds considered by the staff during your documentation of compliance with the ATWS rule (*Code of Federal Regulations* at 10 CFR 50.62.) For your power uprating discuss and justify that the assumptions for the ATWS analysis are adequate as they relate to input parameters such as the initial power level, moderator reactivity feedback, safety relief valves capacity and auxliary feedwater supply. The submittal should include a discussion and applicable values of the unfavorable exposure time, if any, and ATWS core damage frequency.

2. Page 3-10 of Westinghouse Topical Report WCAP-15726 discusses use of the BWCMV-A critical heat flux (CHF) correlation and statistical core design (SCD) methodology for the departure from nucleate boiling ratio (DNBR) reanalysis. The licensee is requested to list the titles of topical reports that document the CHF correlation and SCD methodology, and reference the associated U.S. Nuclear Regulatory Commission (NRC) acceptance letters to confirm the acceptance of the correlation and SCD methodology used in the DNBR reanalysis for power uprate applications. Provide a discussion to address the compliance with each of limitations and restrictions specified in NRC safety evaluations for the applicable topical reports.

3. As stated in the NRC Standard Reivew Plan, one of the acceptance criteria for the transient analysis is related to the calculated DNBR. The staff finds that the information regarding the DNBR reanalysis for power uprate provided in Sections 3.3.7, 3.3.8 and 3.3.9 of WCAP-15726 involves qualitative discussion in nature. The licensee is requested to list the events with calculated DNBRs affected by power uprating and provide calculated results (such as figures showing calculated DNBRs, or margins to DBNR safety limits) for these events to show that the calculated DNBRs for power uprate conditions are within the acceptable safety limits. If the results of reanalysis are more limiting than the analysis of record, the reanalysis results should be included in the updated Final Safety Analysis Report.

4. Stress Corrosion Cracking of Reactor Internals
Increased power is expected to increase the corrosion rates and speed up degradation of reactor internals. Identify the plant programs that are in place to periodically inspect reactor internals and discuss whether these programs are adequate to manage the projected increase of reactor internals degradation due to stress corrosion cracking and primary water stress corrosion cracking.

Enclosure

5. <u>Flow Assisted Cracking (FAC)</u>
Since the effects of FAC on degradation of carbon steel components are plant specific, the licensee needs to provide a predictive analysis methodology which must include the values of the parameters affecting FAC, such as velocity, and temperature before and after the power uprate (PU) and the corresponding changes in components wear rates due to FAC.

6. Indicate the degree of compliance with NRC Generic Letter 89-08, "Erosion/Corrosion in Piping." This letter requires that an effective program be implemented to maintain structural integrity of high-energy carbon steel systems. Describe how this program was modified to account for the PU. If the computer code used in predicting wall thinning by FAC in this program is a generic code (e.g., CHECWORKS), specify it. However, if the code is plant specific provide its description.

7. Identify the predicted change of wear rates calculated by the revised code for the components most susceptible to FAC.

8. Will the PU have significant effect on FAC in balance of plant components? What is the value of the change in FAC wear rates?

9. The response to Question 2 under TXX-99105 (page E8-3 of Enclosure 8) addresses the acceptability of previously performed equipment qualification analyses for a 1.3 percent power increase. Please provide a statement that the previous analyses envelope conditions that will exist after the 1.3 percent power increase, if that is the case.

Mr. J. A. Scalice
Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

cc:
Mr. Karl W. Singer, Senior Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Jon R. Rupert, Vice President (Acting)
Engineering & Technical Services
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Richard T. Purcell
Site Vice President
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

General Counsel
Tennessee Valley Authority
ET 11A
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. Robert J. Adney, General Manager
Nuclear Assurance
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Mark J. Burzynski, Manager
Nuclear Licensing
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Pedro Salas, ManagerLicensing and
Industry Affairs
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

Mr. D. L. Koehl, Plant Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

Mr. Russell A. Gibbs
Senior Resident Inspector
Sequoyah Nuclear Plant
U.S. Nuclear Regulatory Commission
2600 Igou Ferry Road
Soddy Daisy, TN 37379

Mr. Lawrence E. Nanney, Director
Division of Radiological Health
Dept. of Environment & Conservation
Third Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

County Executive
Hamilton County Courthouse
Chattanooga, TN 37402-2801

Ms. Ann P. Harris
341 Swing Loop Road
Rockwood, Tennessee 37854