September 28, 2007

Mr. William R. Campbell, Jr.
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING INCREASED TEMPERATURE AND LEVEL LIMITS OF ULTIMATE HEAT SINK (TAC NOS. MD2621 AND MD2622)

Dear Mr. Campbell:

The Commission has issued the enclosed Amendment No. 317 to Facility Operating License No. DPR-77 and Amendment No. 307 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2. These amendments are in response to your application dated July 12, 2006 (TS-06-03), as supplemented on December 7, 2006, January 26, 2007, May 8, 2007, August 14, 2007, and August 22, 2007.

The amendments revise the technical specifications to establish the minimum water level of the ultimate heat sink (UHS) as 674 feet and to increase the temperature limit of the essential raw cooling water (ERCW) system from 84.5 degrees Fahrenheit (°F) to 87 °F.

The amendments require completion of the following commitments during the implementation process:

a. Eliminate ERCW system flow to nonsafety-related station air compressor loads in the turbine building (Design Change DCN 21523).

b. Rebalance the ERCW system to provide at least 380 gallons per minute (gpm) flow to the shutdown board room chiller.

c. Revise design configuration controls to ensure that the minimum ERCW flow to each emergency diesel generator heat exchanger is at least 400 gpm (including 5 percent measurement uncertainties) and revise calculation MDQ00006720030142 to capture the revised values and the Tubular Exchanger Manufacturers Association references.

d. Revise the Updated Final Safety Analysis Report (UFSAR) to describe the managed river temperature approach. The UFSAR revision should be submitted to the NRC in accordance with the requirements of 10 CFR 71(e).

A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Brendan T. Moroney, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 317 to License No. DPR-77
2. Amendment No. 307 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Brendan T. Moroney, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures: 1. Amendment No. 317 to License No. DPR-77
2. Amendment No. 307 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

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*w/ comments OFFICIAL RECORD COPY

William R. Campbell, Jr.
Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 317
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Tennessee Valley Authority (the licensee) dated July 12, 2006, as supplemented on December 7, 2006, January 26, 2007, May 8, 2007, August 14, 2007, and August 22, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.317, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, and shall be implemented no later than 90 days from the date of its issuance. Implementation shall include completion of the following commitments:

a. Eliminate ERCW system flow to nonsafety-related station air compressor loads in the turbine building (Design Change DCN 21523).

b. Rebalance the ERCW system to provide at least 380 gallons per minute (gpm) flow to the shutdown board room chiller.

c. Revise design configuration controls to ensure that the minimum ERCW flow to each emergency diesel generator heat exchanger is at least 400 gpm (including 5 percent measurement uncertainties) and revise calculation MDQ00006720030142 to capture the revised values and the Tubular Exchanger Manufacturers Association references.

d. Update the UFSAR to describe the managed river temperature approach. The UFSAR revision should be submitted to the NRC in accordance with the requirements of 10 CFR 71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas H. Boyce, Branch Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to License No. DPR-77
and the Technical Specifications

Date of Issuance: September 28, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 317

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace page 3 of Operating License No. DPR-77 with the attached page 3.

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

REMOVE	INSERT
3/4 7-14	3/4 7-14

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis, instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 317, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at power level different from there described; and

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 307
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Tennessee Valley Authority (the licensee) dated July 12, 2006, as supplemented on December 7, 2006, January 26, 2007, May 8, 2007, August 14, 2007, and August 22, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 307, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, and shall be implemented no later than 90 days from the date of its issuance. Implementation shall include completion of the following commitments:

   a. Eliminate ERCW system flow to nonsafety-related station air compressor loads in the turbine building (Design Change DCN 21523).

   b. Rebalance the ERCW system to provide at least 380 gallons per minute (gpm) flow to the shutdown board room chiller.

   c. Revise design configuration controls to ensure that the minimum ERCW flow to each emergency diesel generator heat exchanger is at least 400 gpm (including 5 percent measurement uncertainties) and revise calculation MDQ00006720030142 to capture the revised values and the Tubular Exchanger Manufacturers Association references.

   d. Update the UFSAR to describe the managed river temperature approach. The UFSAR revision should be submitted to the NRC in accordance with the requirements of 10 CFR 71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thopmas H. Boyce, Branch Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Change to License No. DPR-79
and the Technical Specifications

Date of Issuance: September 28, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 307

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace page 3 of Operating License No. DPR-79 with the attached page 3.

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

REMOVE	INSERT
3/4 7-14	3/4 7-14

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 307, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at power level different from there described; and

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 317 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 307 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By application dated July 12, 2006 (Agencywide Document and Management System (ADAMS) Accession No. ML062140102), as supplemented on December 7, 2006 (ML063470029), January 26, 2007 (ML070530142), May 8, 2007 (ML071350246), August 14, 2007 (ML072290326) and August 22, 2007 (ML072360294), hereinafter References (Ref.) 1 through 6, Tennessee Valley Authority (TVA, the licensee) requested changes to the technical specifications (TSs) of the licenses of Sequoyah Nuclear Plant (SQN) Units 1 and 2.

The proposed changes would revise the limiting condition for operation (LCO) of TS 3.7.5, "Ultimate Heat Sink [UHS]." This revision will a) revise the minimum required river water level in TS 3.7.5.a from 670 feet to 674 feet above mean sea level (ft-msl), b) increase the maximum allowed UHS temperature in Section 3.7.5.b from 83 degrees Fahrenheit (°F) to 87 °F, and c) delete the alternate level and temperature limits in TS 3.7.5.c. The licensee has observed that the peak UHS summertime temperature has been slowly increasing over time due primarily to environmental factors, and the proposed changes are deemed necessary in order to support continued plant operation in the event that the current UHS temperature limit is exceeded at some point in the future. The proposed amendments also include administrative changes to delete an expired footnote from a previous amendment and correct a spelling error.

The supplements dated December 7, 2006, January 26, 2007, May 8, 2007, August 14, 2007, and August 22, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 15, 2006 (71 FR 46939).

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act (Act) requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The licensee provides TSs in order to maintain the operational capability of structures, systems and components that are required to protect the health and safety of the public. The regulatory requirements related to the content of the TSs are contained in 10 CFR, Section 50.36. The requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 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; (4) design features; and (5) administrative controls.

In accordance with 10 CFR 50.90, licensees may propose changes to the TSs. In general, there are two classes of changes to TSs: (1) changes needed to reflect modifications to the design bases (TSs are derived from the design bases), and (2) voluntary changes to take advantage of the evolution in policy and guidance as to the required content and preferred format of TSs over time. The proposed amendments deal with the first class of changes, specifically a change to the LCO for the UHS. As stated in 10 CFR 50.36(c)(2)(i), the "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications . . . ."

In determining the acceptability of proposed changes, the U.S. nuclear Regulatory Commission (NRC) staff interprets the requirements of the current version of 10 CFR 50.36, using the accumulation of generically approved guidance, to ensure that the requirements of the plant licensing bases as described in the UFSAR continue to be satisfied. The following sections describe the principal guidance used in the staff's evaluation.

2.1 <u>System Functional Capability</u>

The NRC staff's review of proposed changes in the UHS level and temperature requirements focused on the impact that the proposed changes will have on the capability of structures, systems, and components (SSCs) to perform their assigned safety functions. The criteria that are most applicable to the staff's review of proposed changes to the UHS level and temperature requirements are:

a) 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 5, "Sharing of Structures, Systems, and Components," which specifies that the sharing of SSCs should not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, the orderly shutdown and cooldown of the remaining units; and

b) GDC 44, "Cooling Water," which specifies that a system should be provided to transfer heat from SSCs important to safety to a UHS under normal and accident conditions.

The staff used the guidance provided in Standard Review Plan (SRP) Section 9.2.1, "Station Service Water System," and SRP Section 9.2.5, "Ultimate Heat Sink." Additional NRC requirements and other criteria that apply to SSCs that rely upon the UHS were considered where pertinent. Acceptability of the proposed changes is based primarily upon continued conformance with the plant licensing basis as described in SQN Updated Final Safety Analysis Report (UFSAR) Section 9.2.2, "Essential Raw Cooling Water [ERCW]," and Section 9.2.5, "Ultimate Heat Sink," and as reflected in other UFSAR sections.

2.2 Containment Functional Design

The staff based its review of the containment functional design on the following GDC in 10 CFR Part 50, Appendix A:

a) GDC 16 as it relates to the containment and associated systems establishing a leak-tight barrier against the uncontrolled release of radioactivity to the environment and assuring that the containment design conditions important to safety are not exceeded for as long as the postulated accident require.

b) GDC 38 as it relates to the containment heat removal system safety function which shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident (LOCA) and to maintain them at acceptably low levels.

c) GDC 50 as it relates to the containment heat removal system which shall be designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

The NRC staff also used the following sections of NUREG-0800, "Standard Review Plan," for this review:

a) 6.2.1, "Containment Functional Design"

b) 6.2.1.1.B, "Ice Condenser Containments"

c) 6.2.2, "Containment Heat Removal Systems."

2.3 Loss of Downstream Dam (LODD)

The following GDC were considered in the LODD review:

a) GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that SSCs important to safety shall be designed to withstand the effect of natural phenomena. The SSCs vital to the shutdown capability of the reactor shall be designed to withstand the maximum probable natural phenomenon expected at the site with a sufficient margin.

b) GDC 5, "Sharing of Structures, Systems, and Components," states that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

c) GDC 44, "Cooling Water," requires a system to transfer heat from SSCs important to safety to an UHS. The system shall be capable of performing its safety function under normal and accident conditions.

Regulatory Guide (RG)1.27, "Ultimate Heat Sink for Nuclear Power Plants," provides an acceptable approach for satisfying these criteria. These criteria include recommendations for sufficient cooling ability, integrity during postulated events, function availability and redundancy, and control by the UHS TS. The licensee had evaluated the impacts of the proposed TS changes on the recommended criteria and made a conclusion that these recommended criteria continue to be met. Specifically, the cooling capability of the UHS with the proposed increase in temperature has been evaluated and verified to satisfy the recommendations for heat removal considerations.

The licensee stated that the integrity and availability of the UHS system have not been affected by the proposed TS changes. The licensee stated that the proposed changes to the limiting reservoir elevation levels have been evaluated and verified to continually meet the regulatory requirements for integrity and availability of the UHS. Therefore, the licensee claimed that operation of SQN Units 1 and 2 with the proposed TS changes will not result in a deviation from the recommended criteria in RG 1.27.

3.0 TECHNICAL EVALUATION

3.1 System Functional Capability

This part of the evaluation of the proposed TS changes is based primarily upon the information that was provided by TVA in References 1, 2, 5 and 6. The proposed changes rely in part on a new transient analysis of the UHS for the loss-of-coolant accident (LOCA) coincident with LODD event that was completed by TVA in order to credit more recent information that is available; thereby relaxing some of the conservatism that exists in the current analysis of record.

3.1.1 UHS Transient Analysis Considerations

Evaluation of the proposed TS changes relies to some extent upon the adequacy of the licensee's transient analysis of the UHS following a postulated failure of the downstream dam. In particular, the initial ERCW system (ERCWS) flow rate is based upon the UHS level that is credited for the first 4 hours after the downstream dam fails, and the longer term ERCWS flow rate is based upon the lowest UHS level that remains following the downstream dam failure. The maximum ERCWS supply temperature that is assumed in the licensee's analyses relies upon the capability of the UHS to accommodate the assumed environmental, operational, and accident heat loads over time as the UHS inventory is being reduced due to the downstream dam failure. Relative to these considerations, the licensee provided additional information in response to requests for additional information (RAIs) in Ref. 2 (Response to Questions 1-4, 5, and 7) and Ref. 5 (Response to Question 11).

The NRC staff reviewed the information that was provided for measures that are relied upon by the licensee for preserving the UHS function to confirm that they are adequately described in the UFSAR to assure that they will be maintained in accordance with the plant licensing basis. As discussed in response to Questions 1, 5, 7, and 11(referred to above), the licensee credits river management practices to ensure that the UHS inventory is supplemented with enough upstream river water to prevent the UHS supply temperature from exceeding the maximum allowed value during normal plant operation and postulated accident conditions, including coincident LODD events that result in a reduced UHS inventory. The licensee's response to

Question 11 stated that TVA will continue to use the managed river temperature approach described in TVA document, "Monitoring and Moderating the SQN UHS," which has been used since NRC approval of the last UHS temperature increase to 84.5 °F in 1988. In accordance with the provisions of this procedure, special operations options are implemented in order of severity of the UHS temperature problem that exists. By controlling the timing and quantity of releases from upstream dams, the licensee indicated that TVA has been successful in moderating the summer peak UHS temperatures. For emergency situations, the TVA River Operations Emergency Response Plan specifies steps to follow in the event of a failure of the downstream dam. In response to Question 1 in Ref. 2, the licensee stated that it would update the UFSAR to describe the managed river temperature approach that is credited for preventing the UHS temperature from exceeding the maximum allowed value during implementation of the approved TS changes. The UFSAR revision should be submitted to the NRC in accordance with the requirements of 10 CFR 71(e).

Based upon a review of the information that was provided, the NRC staff considers the licensee's reliance on TVA's river management strategy to be an important consideration for ensuring that the UHS temperature limit for SQN will not be exceeded during normal operating and accident conditions when the summertime temperature is at its peak. The licensee's commitment to describe this strategy in the UFSAR is appropriate and necessary to ensure that the strategy will be properly maintained and implemented as credited in the submittals and plant licensing basis. Therefore, taking the licensee's commitment into consideration, the NRC staff has determined that adequate assurance exists that the UHS will be properly managed and preserved as described by the licensee following implementation of the proposed TS changes and the proposed changes are considered to be acceptable in this regard.

3.1.2 <u>Proposed Change to the UHS Level Requirement</u>

NRC review considerations that could be affected by proposed changes to the minimum required UHS level include the capability of the UHS to provide long-term cooling without exceeding the maximum allowed UHS temperature limit, the impact on ERCWS flow rate assumptions, and the capability to satisfy net positive suction head (NPSH) requirements for the ERCWS pumps. The first item is discussed above in Section 3.1.1, and the other two items are evaluated in this section.

The licensee proposes to establish 674 ft-msl as the minimum required UHS water level. The existing ERCWS flow balance for the component cooling system (CCS) and containment spray system (CSS) heat exchangers is based on a river elevation of 670 ft-msl, which is well below the minimum proposed UHS level of 674 ft-msl. The licensee indicated that a UHS level of 674 ft-msl will supply the ERCWS with cooling water for at least 4 hours before the UHS level drops below 670 ft-msl following a LODD event, which is sufficient to satisfy the short-term cooling demands consistent with the revised heat transfer analysis. Also, with respect to the non-accident unit, the licensee indicated that 4 hours above 670 feet provides enough time for placing the unit in hot standby; thereby maintaining the capability to satisfy the shutdown requirements that are specified by TS limiting conditions for operation.

After the initial 4-hour period following the worst-case LOCA coincident with LODD, the licensee's ERCWS flow balance and corresponding heat removal rate is based on the minimum long-term UHS level of 639 ft-msl that is reached following the downstream dam failure. This compares with the current ERCWS flow balance and heat removal rate that was established based on a minimum long-term UHS level of 639 ft-msl. In Ref. 2 (Response to Question 5), the licensee explained that the minimum UHS water level of 639 ft-msl was established to minimize sediment uptake into the ERCWS. Section 4.2.11.1 of the SQN UFSAR states that an elevation of 639 ft-msl would be reached within about 36 hours after LODD. The 639 ft-msl level will be maintained by releasing sufficient water, 14,000 cfs, through the upstream Watts Bar Dam approximately 12 hours after the LODD occurs. The ERCWS pump sumps are located 14 feet below the 639 ft-msl elevation. Recession curves predict long-term river elevation to be 641 feet at the intake pumping station. Therefore, the licensee's use of 639 ft-msl for the long-term ERCWS flow balance is acceptable.

Based on a review of the information that was provided, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed UHS level requirement on ERCWS flow rate and NPSH considerations. The ERCWS flow rates that are assumed by TVA for short-term and long-term cooling following LOCA coincident with LODD are consistent with those that are currently assumed for the 83 °F UHS temperature limit as approved previously by the NRC in a letter dated August 15, 1988. Also, because a minimum UHS level of 639 ft-msl is assured for long-term cooling, the NPSH for the ERCWS pumps will be increased by three feet which provides additional margin. Therefore, the NRC staff considers the proposed UHS level requirement to be acceptable with respect to ERCWS flow rate and NPSH considerations.

3.1.3 Proposed Change to the UHS Temperature Requirement

NRC review considerations that could be affected by proposed changes to the maximum allowed UHS temperature include the capability of the UHS to provide long-term cooling without exceeding the maximum allowed temperature; containment response; peak fuel cladding temperature; impact on SSCs important to safety; station blackout; environmental qualification of SSCs important to safety; resolution of Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," and other considerations that are identified during the course of the NRC staff's review.

3.1.3.1 Peak Fuel Clad Temperature

The licensee has previously performed an evaluation of the impact that an increase in UHS temperature up to 90 °F will have on the peak fuel cladding temperature. This evaluation was reviewed and approved by the NRC staff in a safety evaluation dated September 30, 2002, for a license amendment requesting an increase in the weight of ice that must be maintained in the containment ice condenser. The proposed increase in the UHS temperature limit to 87 °F is bounded by the previous analysis that was completed and consequently, the peak fuel cladding temperature is not affected by the proposed UHS temperature increase.

The NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed UHS temperature increase on the peak fuel cladding temperature. Because the proposed increase in the UHS temperature limit is bounded by a previous analysis that has been reviewed and approved by the NRC staff, the peak fuel cladding temperature is not affected by the proposed change. Therefore, the proposed increase in the UHS temperature limit is considered to be acceptable with respect to the peak fuel cladding temperature.

3.1.3.2 Impact on SSCs Important to Safety

The licensee's approach for justifying the proposed increase in the maximum allowed UHS temperature is to maintain the existing process cooling water temperatures for heat exchangers served by ERCWS, and cooling water outlet temperatures for SSCs that are cooled by ERCWS, unchanged by taking advantage of excess ERCWS flow margins that currently exist. The available flow margins for individual components are based on existing ERCWS flow modeling that has been developed and validated over time by using flow data that has been gathered during the initial system test program and following system modifications that have been completed. The licensee indicated that the ERCWS flow model is continually updated and validated based on flow data that is gathered following system design changes that are implemented. Based on the ERCWS flow model, the licensee determined where excess ERCWS flow margins were available and sufficient to compensate for the increased UHS temperature such that those SSCs would not be affected by the increased UHS temperature limit. Where sufficient flow margins were not available, the licensee performed more detailed analyses to assess the impact of the higher UHS temperature on the affected SSC in order to determine what additional measures were needed to satisfy design-basis considerations. The NRC staff's review in this evaluation section focuses primarily on those SSCs that have limited excess ERCWS flow margin available or where the basis for acceptance did not appear to be clearly established.

a. <u>SSCs with Limited Excess ERCWS Flow Margins</u>

The analytical approach described above for evaluating the adequacy of existing ERCWS flow margins to compensate for the proposed UHS temperature increase is documented in Calculation MDQ00006720020109, "ERCW System Sensitivity Review for 87 °F." Existing ERCWS flow margins were considered to be adequate for those SSCs that had at least 6 percent more ERCWS flow available than what was required to compensate for the increased UHS temperature; 1 percent more than the 5 percent that was allocated for ERCWS flow measurement uncertainty. SSCs with less than 1 percent flow margin above the 5 percent uncertainty allowance required a more specific evaluation. Additional information pertaining to the licensee's evaluation and those SSCs requiring further evaluation was provided in Ref. 2 (Response to Questions 3, 9, and 10), and Ref. 5 (Response to Questions 12, 13, and 18). The licensee determined that a more detailed evaluation was required for the following components in this regard: the boric acid transfer (BAT) and auxiliary feedwater (AFW) pump room coolers (BAT/AFW CLR 2A and 2B); shutdown board room (SDBR) chillers A and B; and area cooler 1B for the spent fuel pool (SFP) and thermal-barrier booster pump (TBBP).

In order to satisfy the heat removal design criteria for the BAT/AFW room coolers, the licensee credited an increased air flow rate of 14,000 cubic feet per minute (cfm) compared to the previous value of 13,048 (cfm). The licensee confirmed that (i) periodic test results demonstrated that the actual air flow rates exceeded the assumed value of 14,000 cfm, and (ii) the air handler units were originally designed for applications that exceed the 14,000 cfm flow rate that was being credited. Note that the NRC staff considered the second factor to be especially important due to the large uncertainties associated with air flow testing. The licensee's calculations showed that the resulting room temperature would not exceed 117.3 °F, which was slightly higher than the established equipment qualification (EQ) temperature limit of 117 °F for this room. The NRC staff's evaluation of EQ considerations is provided in Section 3.1.3.4.

The second set of components that were within the 1 percent margin criterion are SDBR chillers A and B. The licensee indicated that in order to maintain sufficient ERCWS flow for the SDBR coolers, it would be necessary to eliminate ERCWS flow to the non-safety related station air compressor loads in the turbine building. As committed in Ref. 2 (Response to Question 9), Design Change Notice (DCN) 21523 will be completed to isolate ERCWS flow to the station air compressors, and the turbine building raw cooling water system will be used instead to provide the necessary cooling for the station air compressors. The licensee's evaluation credited some of the additional ERCWS flow margin that was made available by the DCN for increasing the ERCWS flow rates to the SDBR chillers, thereby enabling the SDBR coolers to maintain their respective areas at or below the 100-day post accident average EQ temperature limit. The licensee evaluated the capability of the SDBR chillers to maintain the environment in the main control room in accordance with 10 CFR 50, GDC 19; "Control Room." The analysis showed that there was sufficient margin in the existing ERCWS cooling water flow rate at the limiting UHS temperature of 87 °F to maintain the control room environment without modifying the SDBR chillers. The licensee's ERCWS flow data indicated that the lowest available ERCWS flow rate for the SDBR chillers was 374 gpm, whereas the revised analysis indicated that only 330 gpm was required.

The NRC staff reviewed the vendor data sheets for the SDBR chillers and noted that the original design specification for the ERCWS supply to the SDBR chillers was 85 °F. Because the capacity of the SDBR chillers is reduced with increased cooling water supply temperature, the NRC staff requested that the licensee confirm that the revised ERCWS flow rate analysis was valid for the reduced cooling capacity of the SDBR chillers that exists at the higher UHS temperature limit. Additional information was provided in Ref. 6 to address the NRC staff's concern. After taking the reduced SDBR chiller capacity into consideration, the licensee found that the minimum required ERCWS flow rate for each of the SDBR chillers would be 380 gpm at the increased UHS temperature of 87 °F. However, as indicated above, only 374 gpm is currently available and the licensee indicated that the ERCWS flow rate to the SDBR chillers would be increased to at least 380 gpm by rebalancing a portion ERCWS prior to implementing the proposed increase in the UHS temperature limit. The licensee's commitment, as established in Ref.6, provides adequate assurance that the ERCWS will be properly balanced in accordance with the licensee's revised heat load analysis, thereby ensuring adequate ERCWS flow to the SDBR chillers while still maintaining the required ERCWS flow rates to the other SSCs that are served by ERCWS.

The third and last item that failed to satisfy the 1-percent margin criterion is SFP/TBBP area cooler 1B. The licensee's evaluation indicated that this area cooler had slightly more than one percent ERCWS flow margin over the base 5-percent criterion that was established and, consequently, it was not flagged for a more detailed evaluation. However, the licensee found that there was an error in the evaluation that had been completed and upon reanalysis, found that the available ERCWS flow margin above the 5-percent criterion was actually less than 1 percent for the SFP/TBBP area cooler 1B. Based upon the more detailed evaluation that was subsequently completed, the licensee determined that the area cooler will not exceed its design temperature limit at the higher UHS temperature of 87 °F, but the room temperature will increase during accident conditions. However, the licensee determined that the resulting peak room temperature remains significantly lower than the EQ temperature profile for the room and, hence, the proposed increase in the UHS temperature limit has no adverse effects.

Based upon a review of the information that was provided, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed UHS temperature increase on SSCs with available ERCWS flow margins that are limited. However, by crediting the limited ERCWS flow margins that are available for this equipment, making ERCWS alignment changes as necessary to increase flow margins, and by performing component-specific detailed analyses, the licensee has shown that the SSCs served by the ERCWS that have limited available flow margins will continue to be capable of performing their safety functions in accordance with the plant design basis without exceeding any design limitations (with the noted exception to EQ requirements for the BAT/AFW pump room). In consideration of the licensee's commitments to isolate ERCWS flow to the station air compressors, and to properly balance the ERCWS to provide at least 380 gpm flow to the SDBR chillers while still maintaining the required ERCWS flow balance to the other SSCs that are served by the ERCWS in accordance with the revised heat load analysis that has been completed, the NRC staff finds that SSCs served by the ERCWS will continue to perform their safety functions at the higher UHS temperature limit. Therefore, the proposed increase in the UHS temperature limit is considered to be acceptable with respect to those SSCs that were found to have available ERCWS flow margins that are limited.

b. Auxiliary Feedwater

The ERCWS is relied upon as the seismic Category 1 source of makeup water for the AFW system, and the AFW suction piping temperature design limit is 120 °F. The information that was initially submitted indicated that the ERCWS could provide the AFW pumps with 87 °F water if the condensate storage tanks (CSTs) are not available. The ERCWS makeup supply for the turbine driven AFW pump is taken directly from the ERCWS supply header and the slight increase in UHS temperature does not impact the turbine driven AFW pump suction piping temperature design limit. However, the ERCWS supply for the motor driven AFW pumps is taken downstream of either the CCS heat exchangers or the containment spray heat exchangers, after the ERCWS water has been heated by the respective heat exchanger heat loads, and it was not clear how the ERCWS could supply the motor driven AFW pumps with 87 °F water as indicated. The licensee provided additional information to address this concern in Ref. 2 (Response to Questions 7 and 8), and Ref. 6 (Response to Questions 5 and 16).

The licensee clarified that for the worst-case scenario, ERCWS makeup to the motor driven AFW pumps could reach as high as 128 °F, which exceeds the AFW supply piping design temperature limit of 120 °F. Consequently, SQN analyzed the affected motor driven AFW pump suction piping in accordance with the provisions of the American Society of Mechanical Engineers piping code, Section B31.1, "Power Piping," in order to demonstrate acceptable performance at 128 °F.

Based upon a review of the information that was provided, the NRC staff finds that the licensee has adequately addressed the impact of the proposed UHS temperature increase on the capability of the AFW pumps to perform their safety functions. In particular, the licensee has confirmed that the AFW pump suction piping design limitations will not be exceeded by the higher ERCWS supply temperatures that may occur. Therefore, the NRC staff considers the proposed increase in the UHS temperature limit to be acceptable with respect to AFW makeup capability.

c. CCS Heat Exchangers

Within the SQN site program for implementing the provisions of GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” all safety related heat exchangers that are served by ERCWS are listed and their performance monitoring method and frequency of test are specified. Additional information was provided relative to heat exchanger performance monitoring in Ref. 2 (Response to Question 2), and clarification was provided relative to the description of the CCS heat exchangers in Ref. 6 (Response to Question 2). The licensee indicated that the thermal performance of the CCS heat exchangers is periodically monitored to verify that the heat transfer capability is adequate to meet design basis heat loads. Other heat exchangers receive periodic inspection and maintenance. Although the CCS heat exchanger is a plate heat exchanger (PHE), it is modeled by Westinghouse as a shell and tube heat exchanger in the containment analysis due to software limitations. The licensee confirmed that the CCS plate heat exchangers meet or exceed the heat transfer capability that is assumed in the Westinghouse containment analysis.

The CCS is part of the emergency core cooling system (ECCS) and is designed for all phases of plant operation and shutdown. The CCS is designed to provide cooling water in the temperature range of 35 °F to 95 °F. Calculations 70D530HCGKBO102287, “SQN-CCS Plate Heat Exchangers (PHE) Train 1A/2A ERCW Flow Requirements,” and 70D53EPMMCG021290, “SQN-CCS Plate Heat Exchangers OB1 & OB2 Train B ERCW Flow Requirements,” specify the following design limitations for the CCS heat exchangers: 1) a maximum ERCWS exit temperature of 126 °F for piping design considerations, 2) a maximum CCS return temperature of 145 °F for CCS piping and support design considerations, 3) a maximum CCS exit temperature of 104.5 °F during the LOCA-Recirculation phase for containment pressurization considerations, and 4) a maximum CCS exit temperature of 95 °F for normal plant operation and for refueling heat load considerations. The calculations also indicated that in order to preclude long term fouling during normal operation, the ERCWS flow velocity should be maintained greater than 1.5 ft/sec. This equates to 4330 gpm with two CCS heat exchangers in operation or 2165 gpm with one CCS heat exchanger in operation.

The licensee’s calculations showed that the minimum ERCWS flow rate that is required for PHEs 1A1/1A2 to support one unit in LOCA-Recirculation phase was 3605 gpm; whereas the available flow rate (reduced by 5 percent for measurement uncertainty) was 3671 gpm. The minimum required ERCWS flow rate for PHEs 2A1/2A2 with the opposite unit in hot standby was 1350 gpm; whereas the available flow rate (reduced by 5 percent for measurement uncertainty) was 1601 gpm. The minimum required ERCWS flow rate for swing PHEs OB1/OB2 to support spent fuel pit loads was 3365 gpm; whereas the available flow (reduced by 5 percent for measurement uncertainty) was 5286 gpm. These ERCWS flow rates represented an increase over those that are currently specified in order to compensate for an additional CCS flow tolerance allowance along with the increased UHS temperature of 87 °F in order to prevent PHE inlet and outlet piping temperatures from exceeding the specified limits. However, as discussed in Ref. 2 and Ref. 6 (Response to Questions 8 and 7, respectively), there are circumstances where the heat removal rate has to be controlled by plant operators in order to keep from exceeding the 126 °F and 145 °F criteria. Procedural controls for this purpose are contained in Operations Procedure 0-SO-74-1, “Residual Heat Removal System (RHR),” which directs plant operators to adjust CCS or RHR flow rates accordingly. These procedural controls are part of the normal process for placing RHR in service, and do not represent new procedural requirements.

Based upon a review of the information that was provided, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed UHS temperature increase on the capability of the CCS to perform its safety functions. The licensee has demonstrated that sufficient ERCWS flow margin for the CCS heat exchangers is available to compensate for the proposed temperature increase such that CCS will not be adversely affected. Therefore, the NRC considers the proposed increase in the UHS temperature limit to be acceptable relative to the CCS.

d. Emergency Diesel Generator (EDG) Heat Exchangers

The ERCWS is used to reject heat from the EDG jacket water heat exchangers. The normal EDG jacket water exit temperature (process side) ranges from 165 to 170 °F, and no derating is required if the jacket water exit temperature is kept below 190 °F. Using commercial heat exchanger evaluation software, Tubular Exchanger Manufacturers Association (TEMA) standards, vendor design information, and actual thermal performance data, the licensee developed calculation MDQ00006720030142, "Emergency Diesel Generator (EDG) ERCW Heat Exchanger Evaluation for 87 °F." The licensee determined that the minimum required ERCWS flow rate for the EDG jacket water coolers that was necessary in order to avoid derating ranged from 365 gpm with no tubes plugged to 425 gpm with 5 percent of the tubes plugged. Based upon flow modeling, the licensee found that the available ERCWS flow rate to the ERCWS heat exchangers ranged from 452 gpm for the most limited heat exchanger to 530 gpm. However, the licensee's calculations were based upon jacket water heat exchanger fouling factors in the range of 0.0016 hr-°F-ft^2/BTU compared to the TEMA recommended average fouling factor of 0.0020 hr-°F-ft^2/BTU that applied to SQN.

The licensee provided additional information concerning the most limiting scenario for the EDGs relative to the higher UHS temperature of 87 °F and the fouling factors that were used in the analysis in Ref. 2 (Response to Question 11) and Ref. 6 (Response to Questions 6 and 19). The licensee explained that the lower fouling factors that were used in the evaluation were based upon actual operating conditions and previous plant experience. The NRC staff felt that in the absence of compelling heat exchanger performance data, there was no basis for accepting fouling factors that are less than those recommended by TEMA. In this case, since the EDG heat exchangers are operated infrequently and for relatively short periods of time, it would be very difficult for the licensee to justify the use of the lower fouling factors. Consequently, the licensee performed the calculation again using the TEMA recommended average fouling factor of 0.0020. hr-°F-ft^2/BTU that were applicable for the EDG jacket water heat exchanger flow velocities. With no allowance for tube plugging, the results of this calculation indicated that the minimum required ERCWS flow rate for the jacket water heat exchangers necessary to avoid derating is 400 gpm (includes 5 percent margin for measurement uncertainty). In Ref. 5, the licensee made a regulatory commitment to revise their design configuration controls to ensure that the minimum ERCWS flow rate to each EDG jacket water heat exchanger is at least 400 gpm, and to revise EDG calculation MDQ00006720030142 to reflect the revised fouling factor assumptions and the TEMA reference.

Based upon a review of the information that was provided, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed UHS temperature limit increase on the EDGs. The licensee's analyses show that with a minimum required ERCWS flow rate of 400 gpm to the jacket water heat exchangers, the EDGs will continue to be capable of performing their safety functions following the proposed increase in the UHS temperature limit without exceeding any design limitations. Recognizing the licensee's commitment to ensure that the EDG calculation and design configuration controls are updated to incorporate the analytical results and assumptions as appropriate, the NRC staff finds that the EDGs will remain capable of performing their safety functions at the higher UHS temperature limit.

Therefore, with the design change to ensure at least 400 gpm flow to each EDG jacket water heat exchanger, the proposed increase in the UHS temperature limit is considered to be acceptable with respect to the EDGs.

e. Spent Fuel Pool Cooling

The spent fuel pool (SFP) heat exchangers are cooled by CCS, which is cooled by the ERCWS, for all normal modes of operation and during accident conditions. The design basis in UFSAR Section 9.1.3, "Spent Fuel Pit Cooling System," describes the ability of the SFP cooling system to maintain the SFP at or below 150 °F using two SFP cooling trains. The system consists of two seismic Category 1 cooling trains, each train having a pump and a heat exchanger with a third pump as a reserve. The licensee provided additional information concerning the SFP design and licensing basis in Ref. 2 (Response to Question 6).

As discussed above, the licensee's approach for justifying the proposed increase in the maximum allowed UHS temperature is to maintain the existing process cooling water temperatures for heat exchangers served by ERCWS unchanged by taking advantage of excess ERCWS flow margins that currently exist. The sensitivity analysis referred to above in Section 3.1.3.2.a and the licensee's evaluation of the CCS heat exchangers discussed above in Section 3.1.3.2.c, indicate that sufficient ERCWS flow margin is available to keep the CCS heat exchanger process temperatures from changing. Since the SFP heat exchanger is cooled by CCS and the CCS process temperature is not affected by the proposed increase in the UHS temperature limit, SFP cooling is not affected by the proposed TS change.

Based upon a review of the information that was provided, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed increase in UHS temperature on SFP cooling considerations. Because the CCS heat exchanger process temperatures are not affected by the proposed temperature increase, SFP cooling will not be affected by the proposed change. Therefore, the proposed increase in the UHS temperature limit is considered to be acceptable with respect to SFP cooling.

3.1.3.3 Station Blackout

The licensee indicated that the commitments made to bring SQN into conformance with 10 CFR 50.63, "Loss of All Alternating Current Power," have been fully implemented and continued compliance is not challenged by the proposed UHS temperature increase. The licensee provided additional information concerning the impact of the proposed UHS temperature limit increase on station blackout (SBO) considerations in Ref. 5 (Response to Question 9). Calculation MDQ00006720030142, "Emergency Diesel Generator (EDG) ERCW Heat Exchanger Evaluation for 87 °F," determined the minimum cooling water flow needed for a single EDG to recover from an extended SBO coincident with a loss of all ERCWS event. Recovery from SBO was based in part upon maximum allowable EDG operating and cooling water temperatures with no tube plugging such that engine brake horse power and generating capacity is not derated. The NRC staff's evaluation of the EDG is provided above in Section 3.1.3.2.d. Also, AFW makeup for SBO is provided from the CSTs and not the UHS.

Consequently, the licensee found that SBO mitigation is not affected by the proposed increase in the UHS temperature limit

Based upon a review of the information that was provided, the NRC staff finds that the licensee has adequately addressed the impact of the proposed UHS temperature increase on SBO considerations. Because AFW makeup does not rely upon the UHS during the SBO coping period and ERCWS cooling for the EDGs will continue to be adequate, the proposed increase in UHS temperature to 87 °F will not adversely affect SQN resolution of the SBO issue. Therefore, the NRC staff considers the proposed increase in UHS temperature to be acceptable with respect to SBO considerations.

3.1.3.4 Equipment Qualification (EQ)

The licensee constructed an EQ profile that modeled the short term main steam line break (MSLB) and the long term LOCA temperature profiles. The short term MSLB profile is unaffected by the UHS temperature increase, but the long term heat removal rate is decreased and the containment temperature is marginally increased during the cool down period. As discussed in Section 3.1.3.2.a, the licensee determined that the BAT/AFW pump room temperature could exceed the EQ temperature limit by 0.3 °F. The licensee’s justification for allowing this minor discrepancy is that the overage is very small, and the required heat load, fouling factors, and 10 percent degraded performance factor used in the analysis are collectively very conservative. Also, the UHS temperature is not expected to remain at 87 °F for the entire 100-day period. Therefore, the licensee concluded that the small temperature overage of 0.3 °F is insignificant and an EQ program change is not warranted for this particular situation.

Based upon a review of the information that was provided, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed UHS temperature increase on EQ considerations. The licensee determined that the only area that was affected by the proposed change is the area in the auxiliary building that is cooled by the BAT/AFW pump room coolers. Because the calculated room temperature only exceeds the EQ temperature profile by only 0.3 °F, and recognizing the conservatisms that exist, the NRC staff agrees that the temperature overage is acceptable for this particular situation and that an EQ program change is not warranted. Therefore, the NRC staff considers the proposed increase in the UHS temperature limit to be acceptable relative to EQ considerations.

3.1.3.5 Resolution of GL 96-06

GL 96-06, “Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions,” requested that licensees evaluate the effects of elevated temperature conditions that exist in containment following an accident to assure that (i) thermal over-pressurization of piping systems that penetrate containment will not occur, and (ii) water hammer and two-phase flow conditions in the cooling water systems that serve containment coolers will not compromise plant safety. The licensee indicated that the previous analyses that were completed for addressing the GL 96-06 concerns are not affected by the proposed increase in the UHS temperature limit, and that TVA’s resolution of GL 96-06 will continue to be valid. The NRC staff accepted TVA’s resolution of GL 96-06 in a letter dated April 4, 2000.

Based upon a review of the information provided in the current submittal, the NRC staff finds that the licensee has adequately evaluated and addressed the impact of the proposed UHS temperature increase on TVA's resolution of GL 96-06 for SQN. Therefore, the NRC staff considers the proposed increase in the UHS temperature limit to be acceptable relative to GL 96-06 considerations.

3.1.3.6 Additional Review Considerations

a. Design Control

Enclosure 2 of Ref. 1 discusses how the proposed TS changes will be incorporated into the affected SQN design documents in accordance with the provisions of the SQN design control program. In particular, calculation MDQ00006720020110, "ERCW System Calculations Review for 87 °F," identified all of the design documents that are affected by the proposed UHS temperature increase. The licensee indicated that the affected design documents will be updated as necessary in accordance with Standard Programs and Processes Procedure 9.3, "Plant Modifications and Engineering Change Control."

Based upon a review of the information that was provided, the NRC staff finds that the licensee has adequately considered and addressed the design control requirements that are specified by 10 CFR 50, Appendix B, Criterion 3, "Design Control." In particular, the licensee's design control programmatic controls provide adequate assurance that the affected SQN design control documents have been or will be properly updated as appropriate. Therefore, the NRC staff considers the proposed TS changes to be acceptable with respect to design control considerations

b. ERCWS Flow Modeling and Uncertainty Considerations

As discussed above in Section 3.1.3.2, the licensee developed a computer modeling code to perform a steady-state hydraulic analysis of the ERCWS and found that the system had excess flow capability. The licensee deemed it reasonable to utilize these ERCWS flow margins to offset the higher UHS temperature limit in validating acceptable performance of affected SSCs. In Ref. 5 (Response to Question 3), the licensee explained that their ERCWS flow model is periodically validated by system flow balance testing and continuing inspections that assure that no degraded or non-conforming conditions exist that would invalidate the conclusions of the analyses. A physical ERCWS flow balance was performed in May 1997, an extensive data gathering test was performed in 2002, physical changes to the ERCWS have been incorporated into the flow model and validated, and the flow model is maintained current through continued updates and validation following ERCWS modifications that are made. The hydraulic model was used to analytically determine the flow rates in the ERCWS for meeting plant design basis conditions.

The licensee has relied heavily on the validity of the ERCWS flow model to justify the proposed TS changes, which raised concerns about the uncertainty considerations in the model. The licensee provided additional explanation of how uncertainty considerations were addressed in Ref. 2 (Response to Question 12) and Ref. 5 (Response to Questions 5, 8, 16 and 17). The licensee indicated that the ERCWS flow evaluations included a 5 percent flow measurement uncertainty factor. With respect to UHS temperature measurement, nominal values were used in monitoring of the ERCWS temperatures for compliance with SQN TS surveillance requirements along with temperature averaging to obtain a representative UHS temperature. SQN has a temperature loop measurement uncertainty of +/- 1.16 °F and to assure that the actual UHS surveillance temperature limit will not be exceeded, the main control room ERCWS water temperature indication is offset by a plus 1.5 °F to assure compliance with the UHS TS temperature limit. The licensee implemented this temperature monitoring practice to assure that the UHS temperature measurement is conservative.

The licensee provided additional information in Ref. 5 (Response to Question 4) to explain to what extent ERCWS pump degradation that is allowed by the in-service testing (IST) program can affect the ERCWS flow rates that are assumed. The licensee indicated that the ERCWS flow model uses minimum design values for pump performance in accordance with the IST program criteria. Hence, the flow model provides a conservative estimation of the ERCWS flow rates.

Based upon a review of the information that was provided, the NRC staff finds that the licensee has adequately addressed uncertainty considerations relative to ERCWS flow modeling and UHS temperature measurement. Measures taken to develop and validate the ERCWS flow model and to keep it current provide adequate assurance that ERCWS flow rates will be maintained consistent with design-basis assumptions, and the margin that is applied to the control room ERCWS temperature indication is sufficient to account for measurement uncertainties that exist. Therefore, the NRC staff considers the proposed increase in the proposed UHS temperature to be acceptable relative to the specific measurement uncertainty considerations that were presented.

It should be noted that the staff's review did not include a detailed evaluation of the licensee's determination of measurement uncertainty, because the licensee's measurement uncertainty practices are not affected by the proposed TS changes and are not considered to be a significant factor for this review. However, they may be subject to future NRC inspection activities.

c. TS Bases Considerations

For informational purposes, the licensee provided a draft of the proposed change to the SQN TS Bases Section B3/4.7.5 for the UHS. Although SQN is not a Standard Technical Specifications (STS) plant, the proposed TS Bases are formatted similar to the STS Bases provided in NUREG-1431. The NRC staff noted that the proposed TS Bases referred to the "average" water temperature of the UHS. The STS contains two limiting conditions for operation for UHS temperature. One is a temperature limit that is based on equipment design capability and cannot be exceeded. The other is a lower temperature that can be monitored as a 24-hour average provided that certain criteria are met. The STS provision for a dual temperature limit is based on Technical Specification Task Force Traveler, TSTF-330. Licensees wishing to employ the STS methodology can do so provided the applicable regulatory criteria are satisfied and addressed in a license amendment request. The licensee indicated in Ref. 1 that it was not requesting to adopt TSTF-330, given that it had employed temperature averaging since the issuance of Amendments Nos. 79 and 70, dated August 15, 1988. These amendments increased the UHS temperature limit to 84.5 °F and defined the UHS to be the average of the two ERCW supply headers. The safety evaluation (SE) contained the statement, "This temperature may be averaged over a period of not more than 24 hours." The licensee had interpreted this to allow use of a 24-hour rolling average for temperature monitoring. The NRC staff informed the licensee that this interpretation was incorrect in that the TS clearly established 84.5 °F as a limit that cannot be exceeded, and there was no discussion of 24-hour averaging in the SE. The statement apparently referred to the surveillance requirement to monitor temperature at least once per 24 hours.

In Ref. 5 (Response to Question 10), the licensee indicated that it would no longer use time based averaging. However, the licensee indicated that the reference to temperature averaging will be retained in the TS Bases since TS 3.7.5 defines the UHS temperature as the average temperature of the two ERCWS supply headers.

The proposed TS bases change also credited "sensitivity analyses" for demonstrating that the containment will not be compromised, even under limiting large-break LOCA (LBLOCA) conditions, for UHS temperatures up to and including 90 °F. The NRC staff pointed out that STS Bases do not contain such a discussion, and sensitivity analyses are typically not credited for demonstrating acceptable performance of the containment. In Ref. 5 (Response to Question 10), the licensee indicated that reference to the sensitivity analyses will be eliminated from TS Bases Section B3/4.7.5 relative to containment integrity considerations.

The staff found that the proposed Bases changes provide a summary statement of the bases or reasons for the proposed TS changes as required by 10 CFR 50.36(a) and should be implemented in accordance with the licensee's TS Bases Control Program.

d. Time-Related Licensing Basis Considerations that Could be Affected

During review of the license amendment request, it wasn't clear to what extent time-related licensing-basis criteria were considered and evaluated. Consequently, the NRC staff asked the licensee to identify and discuss any impacts that the proposed increase in UHS temperature limit will have on licensing-basis considerations that specify time-related criteria associated with plant shutdown, cooldown, or accident mitigation, such as the time after plant shutdown to be on RHR cooling, or time specified for reducing containment pressure by half. In Ref. 5 (Response to Question 1), the licensee indicated that the proposed UHS temperature limit increase would not require any changes to existing TS required completion times, and that shutdown requirements, including 10 CFR 50, Appendix R, safe shutdown requirements, are not affected by the proposed change. The licensee did not identify any other time-related considerations that could be affected by the higher UHS temperature limit.

Based on a review of the information that was provided, the NRC staff finds that the licensee has adequately evaluated and addressed to what extent time-related licensing-basis criteria could be affected by the proposed UHS temperature increase. The licensee indicated that TS completion times and 10 CFR 50, Appendix R, were the only items of this nature that were identified, and that they were not affected by the proposed increase in UHS temperature. Therefore, the NRC staff considers the proposed increase in UHS temperature to be acceptable with respect to time-related licensing-basis considerations that could be affected by the proposed change.

3.1.4 Conclusions

The NRC staff has reviewed the licensee's assessment of the impact that the proposed TS changes to establish 674 ft-msl as the minimum required UHS level and 87 °F as the maximum allowed UHS temperature limit will have on SSCs important to safety. As discussed above, the proposed changes will not cause design limitations or functional capabilities of SSCs important to safety to be compromised. Therefore, the staff considers the proposed minimum river water level of 674 ft-msl and maximum allowable UHS temperature of 87 °F to be acceptable.

3.2 Containment Functional Design

SQN Units 1 and 2 are Westinghouse type pressurized water reactors having four primary loops connected in parallel to the reactor vessel. The primary containment is an ice condenser type. The Tennessee River serves as the water source for the UHS to remove the operating and decay heat produced. As described in the SQN UFSAR, one of the safety functions of the UHS is dissipation of residual and auxiliary heat after an accident. The ERCW system draws water directly from the UHS and supplies it as cooling water to the engineered safety feature containment heat removal system components, which are the CSS heat exchangers and the RHR system heat exchangers, during accidents. According to the UFSAR, the ice condenser provides for very rapid absorption of the energy released from the reactor coolant system in the event of a LOCA. The energy is absorbed by condensing steam in a low temperature heat sink, consisting of a suitable quantity of ice permanently stored inside the containment. The ice containment system markedly reduces the peak containment pressure that would otherwise result in the event of a LOCA. The primary function for the CSS is to spray relatively cold water into the containment atmosphere in the event of a LOCA and thereby ensure that the peak pressure reached inside the containment does not exceed the containment design pressure. The CSS supplement the ice condenser until all the ice is melted, approximately 3600 seconds after the LOCA, at which time it and the RHR system become the sole systems for removing heat directly from the containment.

In a letter dated September 12, 2001, the licensee submitted a license amendment request "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Technical Specification (TS) Change No. 01-04, Revised Ice Weight." In Enclosure 4 to this letter, the licensee included the Westinghouse topical report listed herein as Ref.7. This report contained a revised containment analysis based on 87 °F ERCW temperature to the CSS and CCS heat exchangers and the design ERCW flow to the CSS heat exchangers. The NRC approved the license amendment request and issued an SE by a letter dated September 30, 2002. The SQN UFSAR, Section 6.2.1.3, documents Ref. 7 as the current design basis for containment pressure transients. The Ref. 7 containment response analyses utilize a Westinghouse computer model, LOTIC-1, to calculate the long-term peak containment pressure following a LOCA inside containment. The assumptions used by the licensee in the current licensing amendment request are consistent with the assumptions listed in UFSAR Section 6.2.1.3.4.

3.2.1 Containment Response - Short Term

The parameters of interest for the short term containment response are the peak containment pressure and temperature.

The licensee refers to the UFSAR Section 6.2, where it is stated that the peak containment sub-compartment pressure of 15.7 psig is produced within one second due to flow from an instantaneous double-ended guillotine rupture of the reactor coolant system cold leg pipe within its sub-compartment. The increase of UHS temperature from 83 °F to 87 °F does not impact the peak pressure because the UFSAR analysis does not assume heat removal by UHS during this portion of the accident.

The licensee states, as documented in the UFSAR Section 6.2, that the peak containment temperature of 325.5 °F results from a small MSLB at 30 percent reactor thermal power and occurs early in the transient during blowdown from the faulted steam generator. During this period, increase in the containment temperature is mitigated by the ice condenser, the CSS and the passive heat sinks. The CSS is supplied with water from the refueling water storage tank, which is assumed to be at its maximum TS temperature, without assuming any heat removal by the CSS heat exchanger. The mass and energy releases from the faulted steam generator to the containment are terminated by the steam generator dryout after about 30 minutes as per UFSAR Table 6.2.1-37. The ice bed in the ice condenser does not melt out until many hours after a MSLB and continues to remove energy from the containment. By the time switchover of the CSS to the containment sump occurs and heat removal to the UHS begins, temperatures in containment have decreased substantially because of heat removal by the ice condenser. Therefore, peak containment temperature is not affected, because heat rejected to the UHS is not credited in the analyses during the time of the peak containment temperature.

The staff agrees with the licensee's conclusion that the times of interest for the short term containment response are too short to be affected by the change in the UHS temperature and therefore considers the licensee's assessment acceptable.

3.2.2 Containment Response - Long Term

The licensee states that the peak containment pressure is due to an LBLOCA. The results of the analysis show that the maximum calculated peak containment pressure is 11.44 psig at approximately 7000 seconds, which is less than the SQN Units 1 and 2 containment design pressure of 12.0 psig.

For long term containment cooling, the licensee states that the higher total energy release for an LBLOCA is more limiting than the MSLB and, therefore, results in a longer containment cooldown time. The licensee calculated a maximum containment temperature of approximately 235 °F reached within 1 minute of an LBLOCA, which is less than the containment design temperature of 327 °F. The long term temperature profiles resulting from the analyses provided in UFSAR Figures 6.2.1-16 and 6.2.1-17 show that the maximum temperature inside the containment drops to approximately 165 °F at about 28 hours from the start of the event. This licensee performed the above analyses with an ERCW temperature of 87 °F in conjunction with ERCW at its design flow to the CSS heat exchangers.

In a similar license amendment request dated June 20, 1988, the licensee requested an increase of UHS temperature from 83 °F to 84.5 °F, which was approved by NRC on August 15, 1988. In that request, the licensee performed long term containment cooling analyses for a design-basis LOCA in conjunction with a LODD. The licensee assessed a reduction of 7 percent in ERCW design flow due to LODD. In the long term containment analysis with an ERCW temperature of 83 °F, the licensee showed that the effect of 7 percent less ERCW flow on the containment temperature was an increase of 3 °F at two days after the LOCA and LODD. The combined effect of a 7-percent reduction in ERCW flow at the higher temperature of 84.5 °F was an increase in containment temperature by 4.5 °F maximum at 2 days. In its May 8, 2007, response to an RAI for the current license amendment request regarding the effect on long term containment temperature due to reduced ERCW flow caused by LODD, the licensee stated that the same increase (i.e., by about 3 °F after approximately 10,000 seconds from the two events) would be reasonably expected for an ERCW temperature of 87 °F. The staff agrees with the licensee's assessment.

3.2.3 Conclusion

With respect to the criteria identified in Section 2.2, the NRC staff determined that:

(1) GDC 16 is satisfied because the licensee showed that the containment design conditions important to safety are not exceeded during a postulated design-basis LOCA.

(2) GDC 38 is satisfied because the licensee showed that the containment sprays would remove containment heat to reduce containment pressure and temperature rapidly, consistent with the functioning of ERCW system and other associated systems, following design-basis LOCA and would maintain them at acceptably low levels.

(3) GDC 50 is satisfied because the licensee showed that the containment heat removal system is designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from design-basis LOCA.

Therefore, the proposed license amendment is acceptable relative to containment functionality.

3.3 Loss of Downstream Dam

The UHS was designed to perform a principal safety function of dissipating residual and auxiliary heats after a reactor shutdown or after an accident. The design is consistent with the following four regulatory positions in the RG 1.27:

(1) The UHS should be capable of providing sufficient cooling for at least 30 days.

(2) The UHS should be capable of withstanding the effects of the most severe natural phenomena or events, reasonably probable combinations of less severe phenomena or events, and a single failure of man-made structural features;

(3) The UHS should consist of at least two sources of water unless a single source is large enough, and

(4) The TSs for the plant should include actions to be taken in the event that conditions threaten partial loss of the capability of the UHS.

The licensee stated that no physical change is made nor proposed to the capability or capacity of the UHS with the proposed TS change, and that the above regulatory positions will not be affected. However, the proposed TS changes affect the operations of the UHS system by affecting the future temperature and the level of the reservoir. The purpose of increasing the UHS temperature limit is to provide operating leeway in order to avoid potential unnecessary plant shutdowns.

The goal of the staff evaluation is to determine whether the UHS system with the proposed TS changes can provide enough cooling water to the plant after a postulated downstream dam failure. The staff evaluated the licensee's technical analyses and performed additional confirmatory analyses on the postulated dam failure scenario and reservoir recession rate.

3.3.1 Hydrodynamic Modeling

The Chickamauga Dam consists of a concrete dam for power generation and two earth dams on both sides of the river. The licensee assumed that the most-likely severe failure scenario of the Chickamauga Dam is a seismic-induced breach of the 3000-foot southern earth embankment. The licensee performed an analysis of the hydrodynamic effects of the postulated dam failure to ensure that sufficient water levels are maintained in the Chickamauga Reservoir to provide a sufficient source of cooling water to the plant.

The licensee simulated reservoir levels and outflows from the reservoir after the dam failure, and performed sensitivity tests with a range of parameters (e.g., breach width, breach slope, initial water level, upstream inflow) to investigate the margin on the estimates of the reservoir recession rates. The licensee used the Simulated Open Channel Hydraulics (SOCH) computer model for simulating the hydrodynamics of the river system with a postulated dam breach scenario. In Ref. 3, the licensee submitted technical documentation for the model as well as the source code and a set of sample inputs and outputs.

The SOCH model can simulate a number of complex unsteady flow conditions that commonly occur in the Tennessee River and in cases of a dam breach. It solves unsteady flow equations by a finite difference scheme. The model was set up to simulate the flow in the Chickamauga Reservoir with a postulated dam breach scenario. This model was validated with measured data (Garrison et al., 1969; Ref. 11). Based on the review of the algorithm, documentation, and input and output of the model, the staff concluded that the model is acceptable for simulating Chickamauga Reservoir recessions after dam breach. Thus, the staff review focused on the validity of model application.

3.3.2 Licensee's Postulated Dam Failure Scenario

Breach width is the most important parameter in determining the upstream reservoir recession caused by dam failure. The licensee stated that the breach width of 1000 feet used in previous analyses was overly conservative. The licensee pointed out that current industry practice and the U.S. Bureau of Reclamation Report are in agreement in assuming a maximum postulated dam breach width at 5 times the dam height. The southern (left) Chickamauga earthen embankment has the following dimensions:

- Length of southern earth dam: about 3000 feet
- Top elevation: 706 ft-msl
- Bottom elevation: 630 ft-msl
- Dam height at the potential breaching point: 76 feet.

Therefore, the estimated breach width would be 380 feet. Accordingly, the licensee reduced its assumed breach width from 1000 feet to 400 feet. In addition, the licensee assumed the following conservative dam failure scenario for its hydrodynamic analysis:

- Chickamauga Dam fails during a non-flood event;
- The initial reservoir level at the time of failure is 681 ft-msl, which is a median water level in the reservoir;
- An instantaneous dam failure with an assumed breach width occurs; and
- The breach has vertical side slopes extending to the bottom of the dam of 630 ft-msl.

With the above conservative scenario, the licensee simulated the reservoir drawdown after the postulated dam failure. The licensee performed a comprehensive sensitivity analysis by varying breach parameter values to investigate the uncertainties in the estimated recession rate. The result of the sensitivity analysis revealed that an initial water level of the reservoir is very sensitive to the recession rate.

Therefore, staff pointed out that the licensee should use a conservative initial water level in simulating the recession rate. The licensee's response to RAI #5 of Ref. 3 stated that "conservatism for this submittal was built in by only counting the hours during the recession from elevation 674 ft-msl to 670 ft-msl and that a simulation was added by using an initial level of 675 ft-msl which is exceeded 99.9 percent of the time in actual operations." The result of the licensee's re-simulation showed that the total elapsed time from 674 ft-msl to 670 ft-msl for a breach width of 400-feet is about 4.18 hours, which satisfies the 4-hour-above-670-foot limiting condition. However, the licensee's breach width estimates range from 132 feet to 675 feet, which suggested that the use of 400-foot breach width might not be conservative. Therefore, the staff made a confirmatory analysis after determining a conservative initial reservoir level. The results of the staff's analysis are discussed below.

3.3.3 Initial Reservoir Level

The annual operating guide curves for the Chickamauga Reservoir (in Ref. 3) show that the lowest normal operating zone during the wet period (mid-May to August) is 682.5 ft-msl while that of the dry period (December to next March) is 675 ft-msl. However, in the initial application the licensee used a median historical reservoir water level of 681 ft as an initial condition in the simulation of the reservoir recession rates.

To determine a reasonable initial reservoir level, staff requested long-term historical data at the Chickamauga Reservoir. In its response to RAI #6 of Ref. 3, the licensee provided water level and temperature time series. The licensee also confirmed that there was no UHS shutdown experienced during the operation of SQN, implying that there is a leeway in operating the UHS system from the current limiting conditions.

Referring to the attached figures, Figure 1 demonstrates that long-term annual minimum water levels have increased consistently due to the increasing of minimum upstream releases with the revised water control policies in the river. Figure 2 shows that reservoir levels have not dropped below 675 ft-msl since 2000, even during the dry season. Figure 3 shows that initial water levels greater than 675 ft-msl induce a milder recession rate than that for an initial level of 674 ft-msl. After the steep recession during the first 2 hours, the recession rates with the initial level above 675 ft-msl are about 1 foot per hour. In other words, the recession rates with initial water levels above 675 ft-msl automatically may meet the 4-hour-above-670-foot limiting condition.

Therefore, the staff used a minimum reservoir level of 675 ft-msl, which is reasonable and conservative in estimating both breach width and recession rate. Using this initial water level, the staff estimated breach width and time as described below.

3.3.4 Dam Breach Width vs. Recession Time

In general, an earth dam breach starts at the toe of the dam, propagates to the top and develops to the bottom, and continues to the sides of dam as a widening process. Because of the widening process, the reservoir storage volume could be an important parameter in predicting the breach width, especially for a large reservoir like the Chickamauga Reservoir. Many empirical equations are readily available to predict breach parameters using the dam height, the reservoir volume, or both (Ref. 9). Therefore, the staff recommended that the licensee consider the reservoir volume in predicting the breach width. In its response to RAI #1 in Ref. 3, the licensee provided dam breach estimates using four different equations. The resulting breach widths range from 132 feet to 675 feet, from which the licensee assumed the breach width of 400 feet as an average of the estimated breach widths (Ref. 3).

The licensee noted in its response to RAI #2 in Ref. 3 that the small database of large-dam failures tends to indicate 500 feet as a possible upper bound for breach width (page 15 of Ref. 6). However, the staff investigation of the U.S. Bureau of Reclamation database cited by Ref. 9 revealed that 10 out of 410 listed dam breach events exceed a breach width of 500 feet. In the same database, the reported maximum breach width is 5800 feet. Therefore, the staff concluded that the breach width of 400 feet at the Chickamauga Dam is not conservative.

Instead, the staff evaluated breach widths and times using the same four equations as the licensee did but with an initial reservoir level of 675 ft-msl (Table 1). The reason for estimating breach times is to investigate the margin in the reservoir recession rates. The table also includes elapsed times from 674 ft-msl to 670 ft-msl for the estimated breach width equations. The elapsed times were computed based on a relationship between breach width and elapsed time given in Figure 4 (attached), which was constructed by the licensee's reservoir recession rates simulated with the SOCH model.

Table 1. Staff estimates of breach width, breach time, and time elapsed from 674 ft-msl to 670 ft-msl, with an initial reservoir level of 675 ft-msl, initial reservoir storage volume of 392,000 acre-feet, and upstream inflow of 14,000 cubic feet per second.

Equation	Estimated Breach Width (foot)	Estimated Breach Time (Hours)	Time Elapsed 674-670 ft-msl (Hours)
Von Thun and Gillette (1990)	297	0.20	4.37
U.S. Bureau of Reclamation	135	0.45	4.70
Federal Energy Regulatory Commission	380	1.27	4.21
Froehlich (1995)	645	6.02	3.68

It should be noted that the staff's breach width estimates are nearly identical to those of the licensee. The first three equations in the table use the breach height as an independent variable while the Froehlich equation uses both the breach width and the storage volume as independent variables.

The result shows that the estimated breach widths and times vary noticeably from equation to equation. The breach time by the Froehlich equation is about 30 times longer than the minimum. The elapsed times of the first three equations meet the 4-hour-above-670-foot limiting condition, while that of the Froehlich equation violates the limiting condition. However, the breach time by the Froehlich equation is substantially longer than those of the other three equations, implying that the actual recession time from 674 ft-msl to 670 ft-msl by the Froehlich equation will be delayed substantially if the simulation of the reservoir drawdown considers the estimated breach time instead of assuming an instantaneous breaching. Therefore, the staff concluded that the UHS system with the proposed TS changes meets the 4-hour-above-670-foot limiting condition.

3.3.5 Long-term Recessions

The licensee performed a comprehensive sensitivity analysis with the SOCH model to investigate a variability of recession rates with a range of potential breach parameters and varying upstream inflows. The licensee found that the wider breach (W=1000 feet) results in recession times longer than that of shorter breach width (W=300 feet), but the initial ($t<12$ hours) and final ($t>60$ hours) recessions of two different breach widths are nearly identical. Based on the results of the sensitivity tests, the licensee concluded that the breach parameters are not sensitive to the recession rate, especially at the beginning and end of the recession.

Using the SOCH model, the licensee developed a family of recession curves for different inflow rates from the Watts Bar Reservoir into the upstream of the Chickamauga Reservoir, from which they developed an outflow rating curve. This new model-generated rating curve would slightly lag the original 1988 prediction recession. The maximum delay is 2 to 3 hours, occurring between 24 and 36 hours after dam failure. The licensee stated that using model-generated curves would predict a slightly lower steady-state elevation after 60 hours.

Simulation results by the licensee indicate that after 60 hours past dam failure, reservoir levels would be maintained in a steady-state mode and the levels are entirely dependent on the river geometry and the inflow from the upstream Watts Bar Dam. The SQN UFSAR states that a minimum discharge into the Chickamauga Reservoir is 14,000 cfs. With this inflow volume, the steady-state reservoir level reaches about 641 ft-msl, which is marginally higher than the requested long-term UHS level of 639 ft-msl. Based on the information provided, the staff concluded:

- Under a postulated dam break condition, the proposed maximum UHS water level change will not violate the provision of a 4-hour-above-670-foot limiting condition, i.e., the proposed minimum UHS intake water level is maintained under the proposed UHS maximum water level change with a postulated dam break scenario.

- The result of a hydrologic simulation provided by the licensee demonstrates that, with an upstream inflow of 14,000 cubic feet per second, a steady reservoir level of 641 ft-msl is maintained after dam break, satisfying the minimum UHS intake level to maintain a sufficient UHS pump suction head.

- The temperatures at the discharge outlet and reservoir will not increase by the proposed UHS temperature change, mainly because of the excess UHS cooling water.

Therefore, the UHS system with the proposed TS changes meets the requirement for long-term cooling.

3.3.6 Margin in the Estimated Recession Rate

The staff asked whether there are sufficient margins to ensure at least 4 hours of river level above 670 ft-msl following a loss of the downstream dam. The licensee responded (RAI #2 in Ref. 4) that its postulated breach condition (W=400-feet, initial level of 675 ft-msl) meets the 4-hour-above-670-feet limiting condition. The staff's confirmatory analysis shows that the limiting condition is satisfied. Even with the worst case dam breach scenario (by the Froehlich equation), where the breach width exceeds 400-feet, the corresponding breach time is long enough to meet the 4-hour-above-670-foot limiting condition. The margin in the reservoir recession rate is also increased by the following two facts:

First, the submerged weir placed in the Chickamauga Reservoir was not considered in the licensee's hydrodynamic modeling. The weir will delay the drawdown of the reservoir after dam failure. During the construction of SQN, an underwater rock dam was placed in the main channel below the cooling water intake and upstream of the discharge diffuser pipes to help maintain an available pool of cooler water during the summer time. According to the plan drawings of SQN, the underwater dam has the following dimensions:

- A maximum height of 19-feet above the streambed
- A top elevation of 654 ft-msl, and
- Natural upstream/downstream side-slopes

The dam was constructed of quarry rock placed by a bottom-dump barge. The licensee stated that both the current integrity of the structure and its durability under weir flow conditions are unknown. Therefore, the licensee did not consider the impact of the underwater weir in their evaluation. Thus, the river level will be maintained without the weir. If the weir is intact, it will play a role in reducing the rate of reservoir drawdown and provide additional assurance that the level will be maintained.

Second, the implementation of a series of Water Control Rules in the Tennessee River will increase the minimum reservoir levels over time. In other words, the new rule will reduce the chance of the reservoir levels being below 675 ft-msl and will increase the margin in the estimated recession rate. The licensee stated that it adopted the latest Reservoir Operating Policy in 2004 in response to the changing demands on the reservoir systems in the Tennessee River. Significant changes in the policy as they impact SQN included extending the summer water level on the Chickamauga Reservoir until Labor Day and adopting a tiered minimum flow regime from June through Labor Day at the Chickamauga Dam. This requires increasing the minimum average weekly flow requirements in early June to August, except in the driest of years when the flow requirement is only 25,000 cfs beginning August 1. Winter operating levels were also increased on many of the upstream tributary projects, providing more carry-over reservoir storage that can eventually help increase the minimum water levels in the reservoir.

In summary, both the submerged weir and the current water management rule will contribute to delaying the reservoir recession after dam failure and thus enhance the UHS operating margin.

3.4 Administrative TS Changes

The proposed amendments include two administrative changes. The first is to remove a footnote that reads, "*87 °F is allowed until September 30, 1995." This was added by previous amendments to allow a limited duration increase in the UHS temperature, which has since expired. The footnote is no longer applicable and can be removed.

The second is to correct the spelling of the word "Requirements" in the heading of the Surveillance Requirements section. This is also acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official, Mr. Bruce House of the Tennessee Bureau of Radiological Health, was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (71 FR 46939). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. TVA letter to NRC dated July 12, 2006, “Sequoyah Nuclear Plant (SQN) – Units 1 and 2 – Technical Specifications (TS) Change 06-03 ‘Ultimate heat Sink (UHS) Temperature Increase and Elevation Changes.’”

2. TVA Letter to NRC dated December 7, 2006, “Sequoyah Nuclear Plant (SQN) – Units 1 and 2 – Technical Specifications (TS) Change 06-03 ‘Ultimate heat Sink (UHS) Temperature Increase and Elevation Changes, Supplemental Information’ (TAC Nos. MD2621 and MD2622).”

3. TVA Letter to NRC dated January 26, 2007, “Sequoyah Nuclear Plant (SQN) – Units 1 and 2 – Response to Request for Additional Information (RAI) for Technical Specifications (TS) Change 06-03 (TAC Nos. MD2621 and MD2622).”

4. TVA Letter to NRC dated May 8, 2006, “Sequoyah Nuclear Plant (SQN) – Units 1 and 2 – Technical Specifications (TS) Change 06-03 ‘Ultimate Heat Sink (UHS) Temperature Increase and Elevation Changes, Supplemental Information No. 2’ (TAC Nos. MD2621 and MD2622).”

5. TVA Letter to NRC dated August 14, 2007, “Sequoyah Nuclear Plant (SQN) – Units 1 and 2 – Technical Specifications (TS) Change 06-03 ‘Ultimate Heat Sink (UHS) Temperature Increase and Elevation Changes - Request for Additional Information (RAI) No. 2' (TAC Nos. MD2621 and MD2622).”

6. TVA Letter to NRC dated August 22, 2007, “Sequoyah Nuclear Plant (SQN) – Units 1 and 2 – Technical Specifications (TS) Change 06-03 ‘Ultimate Heat Sink (UHS) Temperature Increase and Elevation Changes- Request for Additional Information (RAI) No. 2 Supplement’ (TAC Nos. MD2621 and MD2622).’”

7. Thompson, C.M., Kolano, J.A, and Smith, L.C., “Tennessee Valley Authority Sequoyah Nuclear Plant, Units 1 and 2 - Containment Integrity Reanalyses Engineering Report,” WCAP-12455, Revision 1, Supplement 1R, September 2001.

8. “Updated Predictions of Chickamauga Reservoir Recession Resulting from Postulated Failure of the South Embankment at Chickamauga Dam,” June 2004, Tennessee Valley Authority – River System Operation and Environment, River Operations and River Scheduling (attached as an appendix of Ref. 1).

9. “Prediction of Embankment Dam Breach Parameters,” Report DSO-98-004, U.S. Department of Interior, July 1998.

10. Wahl, T. L., "Uncertainty of Predictions of Embankment Dam Breach Parameters," American Society of Civil Engineers (ASCE) Journal of Hydraulic Engineering, Vol. 130, No. 5, May 1, 2004.

11. Garrison, J.M., J.P. Granju, and J.T. Price, "Unsteady Flow Simulation in Rivers and Reservoirs," Journal of the Hydraulics Division, ASCE, Vol. 95, #HY5, Proceeding Paper 6771, September 1969, pages 1559-1576.

12. Von Thun, J. Lawrence, and David R. Gillette, 1990, Guidance on Breach Parameters, unpublished internal documentation, U.S. Bureau of Reclamation, Denver, Colorado, March 13, 1990, 17 pages.

13. Froehlich, David C., 1995, Peak Outflow from Breached Embankment Dam, Journal of Water Resources and Management, vol. 121, no. 1, p. 90-97.

Attachment: Figures 1 - 4

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Date: September 28, 2007

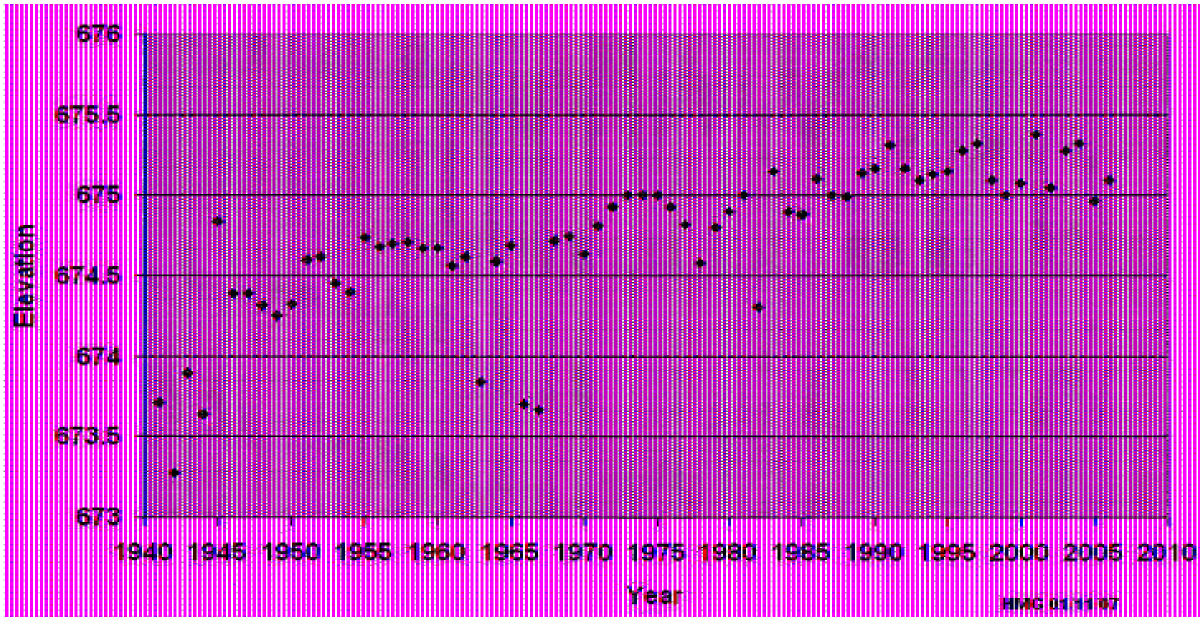

Figure 1. Annual minimum headwater levels (ft-msl) in the Chickamauga Dam (from the licensee, Ref. 3).

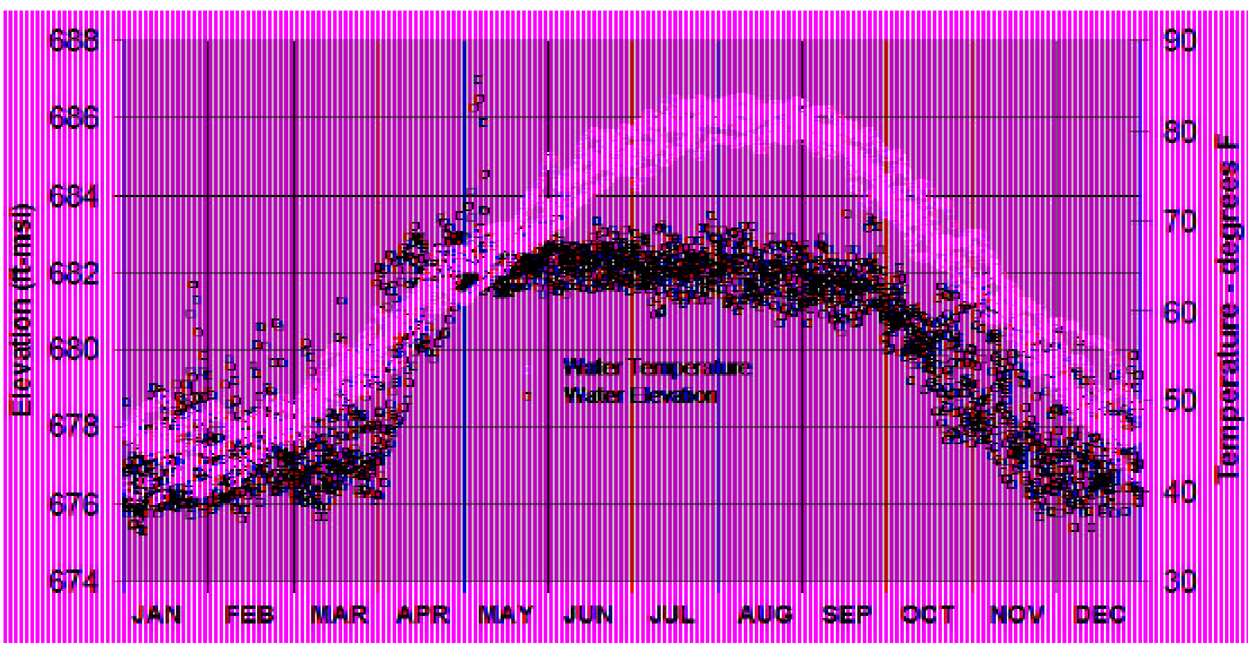

Figure 2. Seasonal headwater levels and water temperatures from 2000 to 20006 at the Chickamauga Reservoir (from the licensee, Ref. 3).

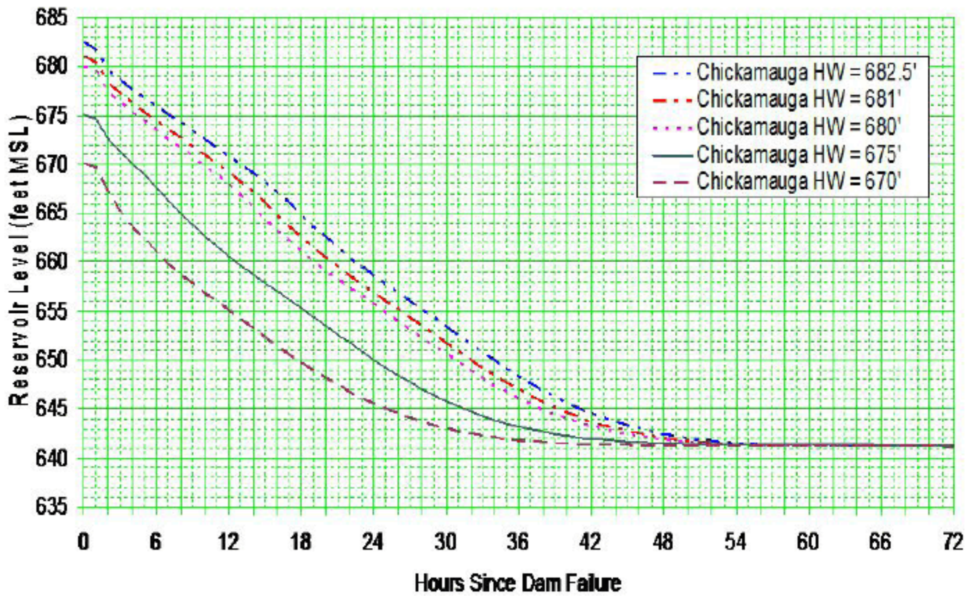

Figure 3. Chickamauga Reservoir recession curves simulated by the SOCH model with different initial reservoir levels, using a breach width of 1000-ft, an initial reservoir level of 630 ft-msl, and a Watts Bar release of 14,000 cfs.

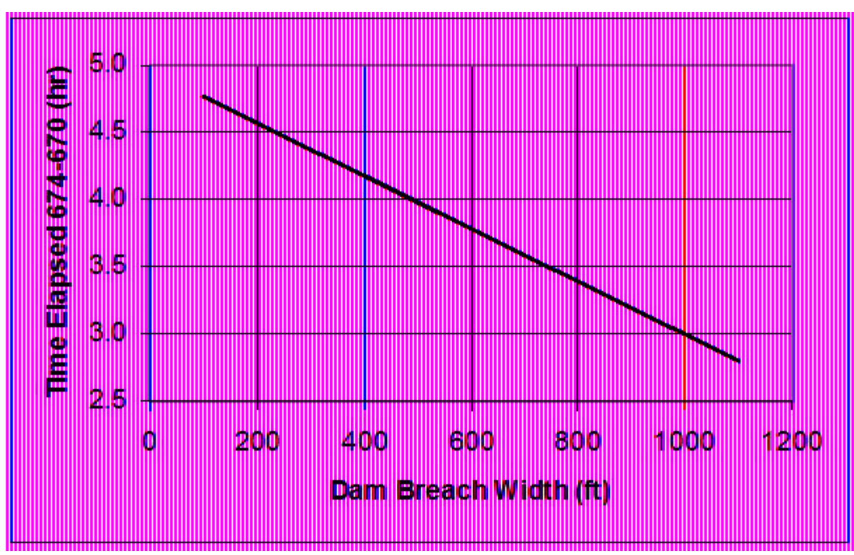

Figure 4. Chickamauga Dam breach width versus elapsed time of the reservoir recession level from 674 ft-msl to 670 ft-msl. This plot was re-constructed based on the licensee's simulations of recession rates using the SOCH model with an initial reservation level of 675 ft-msl (Ref. 3).