September 15, 2004

Mr. Karl W. Singer
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 AMENDMENT REGARDING TECHNICAL SPECIFICATION CHANGE NO. 00-14, PRESSURE TEMPERATURE LIMITS REPORT (TAC NOS. MB6436 AND MB6437)

Dear Mr. Singer:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 294 to Facility Operating License No. DPR-77 and Amendment No. 284 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively.

This amendment is in response to your application dated September 6, 2002, as supplemented by letters dated December 19, 2002, March 28, June 24, September 3, and October 22, 2003. The Tennessee Valley Authority (TVA) submitted a request to revise selected technical specifications (TSs) for SQN, Units 1 and 2. The requested changes included (1) the relocation of the pressure temperature (P/T) limit curves and low temperature over pressure protection system limits to the Pressure and Temperature Limits Report (PTLR), (2) the referencing of the PTLR in the affected TS limiting conditions for operation and bases, including the addition of the PTLR to the definitions section of the TSs, and the addition of a new TS 6.9.1.15 to the administrative controls section of the TSs, (3) the relocation of TS 3/4.4.9.2, Pressurizer, to the SQN Technical Requirements Manual, and (4) the revision of TS 3/4.4.9.1, Pressure/Temperature Limits, Reactor Coolant System, and TS 3/4.4.12, Low Temperature Over Pressure Protection Systems, to incorporate standard TS requirements from NUREG-1431, Revision 2. Based on our review, the Commission has found that the proposed revisions to the SQN TSs are acceptable.

Additionally, TVA requested two exemptions in these applications. The first proposed exemption requested the use of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI Code Case N-640, “Alternative Requirement Fracture Toughness for Development of P/T Limit Curves for ASME Section XI, Division 1” as the basis for the revised reactor pressure vessel pressure-temperature limit curves. The second exemption requested the use of Westinghouse Report WCAP-15984, Revision 1, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2” in lieu of Title 10, *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, for determining the reactor pressure vessel flange minimum temperature requirements.

The first exemption for SQN, Unit 2 was previously approved via separate correspondence (ADAMS Accession No. ML032060558) dated July 30, 2003. This same exemption request is no longer required to be approved by the NRC since Table 1 of Regulatory Guide 1.147, Revision 13 (January 2004), lists N-640, "Alternate Reference Fracture Toughness for Development of P/T Limit Curves, Section XI, Division 1" as acceptable to the NRC for application in licensees' ASME Section XI Inservice Inspection Programs. This Regulatory Guide is approved for licensee use by reference in 10 CFR 50.55a(b). The second exemption was approved for SQN, Units 1 and 2 via separate NRC correspondence (ADAMS Accession No. ML041940552) dated July 7, 2004 .

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/

Robert J. Pascarelli, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 294 to License No. DPR-77
2. Amendment No. 284 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

The first exemption for SQN, Unit 2 was previously approved via separate correspondence (ADAMS Accession No. ML032060558) dated July 30, 2003. This same exemption request is no longer required to be approved by the NRC since Table 1 of Regulatory Guide 1.147, Revision 13 (January 2004), lists N-640, "Alternate Reference Fracture Toughness for Development of P/T Limit Curves, Section XI, Division 1" as acceptable to the NRC for application in licensees' ASME Section XI Inservice Inspection Programs. This Regulatory Guide is approved for licensee use by reference in 10 CFR 50.55a(b). The second exemption was approved for SQN, Units 1 and 2 via separate NRC correspondence (ADAMS Accession No. ML041940552) dated July 7, 2004.

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/

Robert J. Pascarelli, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 294 to License No. DPR-77
2. Amendment No. 284 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page
DISTRIBUTION: See next page

* Concurrence Per Safety Evaluation Input

Package No.: Enclosure:
ADAMS ACCESSION NO. ML042600465 NRR-058

OFFICE	PDII-2/PM	PDII-2/LA	OGC	SRXB/SC	EEIB/SC
NAME	RPascarelli	BClayton	SLewis(NLO)	JUhle*	WBateman*
DATE	08/ /04	08/ /04	9/10/04	10/30/03	08/10/04
OFFICE	IROB/SC	PDII-2/SC (A)			
NAME	TBoyce*	MMarshall			
DATE	08/02/04	9/15/04			

OFFICIAL RECORD COPY

DISTRIBUTION FOR: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 — ISSUANCE OF AMENDMENT REGARDING TECHNICAL SPECIFICATION CHANGE NO. 00-14, PRESSURE TEMPERATURE LIMITS REPORT (TAC NOS. MB6436 AND MB6437)

<u>DISTRIBUTION</u>:
PUBLIC
PDII-2 R/F
RidsNrrDlpmDpr
RidsNrrDlpmpdii (EHackett)
RidsNrrDlpmLpdii2 (MMarshall)
RidsNrrPMRPascarelli
RidsNrrLABClayton (Hard Copy)
RidsRgn2MailCenter (SCahill)
RidsAcrsAcnwMailCenter
GHill (4 hard copies)
RidsOgcRp
TBoyce
JUhle
WBateman
KKavanaugh
MMitchell
SPeters

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

Mr. James E. Maddox, Vice President
Engineering & Technical Services
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Randy Douet
Site Vice President
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37384-2000

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

Mr. John C. Fornicola, Manager
Nuclear Assurance and Licensing
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

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

Mr. Pedro Salas, Manager
Licensing and Industry Affairs
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37384-2000

Mr. David A. Kulisek, Plant Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37384-2000

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

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

County Mayor
Hamilton County Courthouse
Chattanooga, TN 37402-2801

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 294
License No. DPR-77

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

    A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 6, 2002, as supplemented by letters dated December 19, 2002, March 28, June 24, September 3, and October 22, 2003, 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 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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 294, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented by the completion of the 2004 Sequoyah Unit 1 Cycle 13 Refueling Outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael L. Marshall, Jr., Acting Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 15, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 294

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace the following pages of the Appendix A and B Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
I	I
II	II
VI	VI
XIII	XIII
1-5	1-5
1-6	1-6
1-7	1-7
3/4 4-23	3/4 4-23
3/4 4-24	3/4 4-24
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	-----------
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	B 3/4 4-12
B 3/4 4-13	B 3/4 4-13
B 3/4 4-14	B 3/4 4-14
-------------	B 3/4 4-15
-------------	B 3/4 4-16
-------------	B 3/4 4-17
-------------	B 3/4 4-18
-------------	B 3/4 4-19
-------------	B 3/4 4-20
-------------	B 3/4 4-21
-------------	B 3/4 4-22
-------------	B 3/4 4-23
6-13a	6-13a

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 284
License No. DPR-79

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 6, 2002, as supplemented by letters dated December 19, 2002, March 28, June 24, September 3, and October 22, 2003, 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 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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 284, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented by the completion of the 2005 Sequoyah Unit 2 Cycle 13 Refueling Outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael L. Marshall, Jr., Acting Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 15, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 284

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace the following pages of the Appendix A and B Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
I	I
II	II
VI	VI
XIII	XIII
1-5	1-5
1-6	1-6
1-7	1-7
1-8	1-8
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-34	3/4 4-34
3/4 4-35	3/4 4-35
3/4 4-36	-----------
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	B 3/4 4-12
B 3/4 4-13	B 3/4 4-13
B 3/4 4-14	B 3/4 4-14
B 3/4 4-15	B 3/4 4-15
-------------	B 3/4 4-16
-------------	B 3/4 4-17
-------------	B 3/4 4-18
-------------	B 3/4 4-19
-------------	B 3/4 4-20
-------------	B 3/4 4-21
-------------	B 3/4 4-22
-------------	B 3/4 4-23
-------------	B 3/4 4-24
6-14	6-14

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED

TO AMENDMENT NO. 294 TO FACILITY OPERATING LICENSE NO. DPR-77 AND

AMENDMENT NO. 284 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 letter dated September 6, 2002 (Reference 1), and supplemented by letters dated December 19, 2002, March 28, June 24, September 3, and October 22, 2003 (References 2, 3, 4, 5, and 6), the Tennessee Valley Authority (TVA) submitted a request to revise selected technical specifications (TS) for Sequoyah (SQN), Units 1 and 2. The requested changes included (1) the relocation of the pressure temperature (P/T) limit curves and low temperature over pressure protection (LTOP) system limits to the Pressure and Temperature Limits Report (PTLR), (2) the referencing of the PTLR in the affected TS limiting conditions for operation (LCOs) and bases, including the addition of the PTLR to the definitions section of the TSs, and the addition of a new TS 6.9.1.15 to the administrative controls section of the TSs, (3) the relocation of TS 3.4.9.2, Pressurizer, to the SQN Technical Requirements Manual (TRM) and (4) the revision of TS 3.4.9.1, Pressure/Temperature Limits, Reactor Coolant System, and TS 3.4.12, Low Temperature Over Pressure Protection Systems, to incorporate standard TS (STS) requirements from NUREG-1431, Revision 2, "Standard Technical Specifications - Westinghouse Plants." The supplemental letters provided clarifying information that did not expand the scope of the original application or change the initial proposed no significant hazards consideration determination.

Additionally, TVA requested two exemptions in these applications. The first proposed exemption requested the use of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI Code Case N-640, "Alternative Reference Fracture Toughness for Development of P/T [pressure-temperature] Limit Curves for ASME Section XI, Division 1" as the basis for the revised reactor pressure vessel pressure-temperature limit curves. The second exemption requested the use of Westinghouse Report WCAP-15984 Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2" in lieu of Title 10, *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, for determining the reactor pressure vessel flange minimum temperature requirements.

The first exemption for SQN, Unit 2 was previously approved via separate correspondence (ADAMS Accession No. ML032060558) dated July 30, 2003. This same exemption request is no longer required to be approved by the U.S. Nuclear Regulatory Commission (NRC) as Table 1 of Regulatory Guide (RG) 1.147, Revision 13 (January 2004) lists N-640, "Alternate

Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1" as acceptable to the NRC for application in licensees' ASME Section XI Inservice Inspection Programs. This RG is approved for licensee use by reference in 10 CFR 50.55a(b). The second exemption was approved for SQN, Units 1 and 2 via separate NRC correspondence (ADAMS Accession No. ML041940552) dated July 7, 2004.

2.0 REGULATORY EVALUATION

This amendment request has been evaluated from several aspects, each having unique regulatory requirements. The evaluation of these requirements is included in the appropriate section of the Technical Evaluation.

3.0 TECHNICAL EVALUATION

3.1 Fluence Calculations for the PTLR and the Pressure Relief Capacity for the LTOP System

3.1.1 Regulatory Evaluation

A basic assumption for light-water-cooled power reactors is that the reactor pressure vessel (RPV) does not fail. The NRC established reactor design requirements to help prevent this type of failure. General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary," requires that the pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires that the pressure boundary materials be designed, fabricated, erected, and tested to the highest quality standards practical. Additionally, GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires that the pressure boundary be designed with sufficient margin to assure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. Furthermore, GDC 31 requires the design to reflect consideration of service temperatures and other conditions in determining the material properties, stresses, size of flaws, and effects of irradiation on the material properties.

Also, to help prevent vessel failure, the NRC established specific fracture toughness requirements for normal operation and anticipated operational occurrences. As stated in 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors Operation," licensees are required to follow 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." Additionally, in response to concerns over potential pressurized thermal shock (PTS) events in pressurized-water reactors (PWRs), the NRC issued 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

Appendix G describes the establishment of P/T limits for the reactor coolant pressure boundary materials. As an alternative to the 10 CFR Part 50, Appendix G, paragraph IV.A.2.b requirements for P/T limit curve development, TVA proposed using the ASME Boiler and Pressure Vessel Code, Section XI Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P/T limit Curves for ASME Section XI, Division 1," requirements. The P/T limits derived from these methods provide margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary during normal operation, anticipated operational occurrences, and system hydrostatic tests.

However, as exposure to neutron fluence increases, an increasing nil ductility reference temperature (RT_{NDT}) displays the embrittling effect of the fluence on the material toughness of the vessel. The operating P/T limit curves must be adjusted, as necessary, based on the exposure to this neutron fluence. Therefore, to satisfy the requirements of Appendix G, Code Case N-640, and 10 CFR 50.61, application of methods for determining the fast neutron fluence ($E > 1$ MeV) are necessary to estimate the fracture toughness of the RPV materials.

During low temperature operation, the LTOP system controls reactor coolant system (RCS) pressure so the integrity of the reactor coolant pressure boundary is not compromised by violating the P/T limits of 10 CFR Part 50, Appendix G, or those of Code Case N-640. The reactor vessel is the limiting reactor coolant pressure boundary component for demonstrating such protection. The TSs provide the maximum allowable actuation logic setpoints for the power operated relief valves (PORV) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Appendix G or ASME Code Case N-640 requirements during the LTOP modes.

By Generic Letter (GL) 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits” (Reference 7), the NRC established guidance by which to relocate the plants P/T curves and LTOP setpoints from the TSs to a licensee controlled document. Typically, licensees call this document the Pressure Temperature Limits Report. The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic testing, and data for the maximum rate of change of the reactor coolant temperature.

An acceptable method for calculating the P/T curves and LTOP setpoints is WCAP-14040-NP-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves” (Reference 8). Furthermore, RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” (reference 9), describes methods that ensure accuracy and reliability of the fluence determinations required by GDC 14, GDC 30, GDC 31, 10 CFR 50.61, and 10 CFR Part 50, Appendix G.

For this review, the staff reviewed the proposed vessel fluence calculations and LTOP setpoints for compliance with the provisions of GL 96-03, WCAP-14040-NP-A, and RG 1.190.

3.1.2 Technical Evaluation

3.1.2.1 Sequoyah Unit 1

The proposed PTLR is included as Enclosure 4 of Reference 1. The licensee based the update to these limits and setpoints on the Westinghouse topical report, WCAP-15293 (Reference 10). Both the P/T limits and the LTOP setpoints were calculated using the methodology described in WCAP-14040-NP-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” (reference 8), as supplemented by: (1) the ASME Code Case N-640 “Alternate Reference Fracture Toughness for Development of P/T Limits for Section XI, Division 1,” (2) the 1996 version of ASME Section XI, Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” and (3) WCAP-15315, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR [Boiling-Water Reactor] Plants” (Reference 11).

3.1.2.1.1 WCAP-15293, Revision 1, “Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation”

3.1.2.1.1.1 Appendix C, “Calculated Fluence Data”

The licensee developed the best estimate exposure for the SQN, Unit 1 reactor vessel in WCAP-15224 (Reference 12), using a combination of absolute plant specific transport calculations and plant specific measurement data.

The staff found the pressure vessel projected fluence values to be acceptable because the calculation was based on the DORT code, a 2-D discrete ordinates in the (r,θ) and (r, z) planes. Additionally: (1) the methodology is consistent with reference 9, (2) the code approximations (i.e., P3 for the anisotropic scattering expansion and S_8 for the angular quadrature) are acceptable, (3) the cross section file (BUGLE-96) was based on ENDF/B-VI, which is recommended in Reference 9, and (4) the results indicate that the M/C ratio is 1.14 for E>1.0 MeV, which is conservative with respect to the calculated value.

3.1.2.1.2 Low Temperature Overpressure Protection System (LTOPS) Setpoints (TS 3/4.4.12)

The pressurizer power operated relief valve (PORV) lift setpoints are determined from the P/T curves and satisfy 10 CFR Part 50, Appendix G. The setpoints do not include instrument uncertainties, however, TVA quantified the instrument channel uncertainties and evaluated the setpoints versus the limits of 10 CFR Part 50, Appendix G. The calculated enable temperature could be based on the ASME Code Case N-514 requirements. These would allow SQN, Unit 1 to use RT_{NDT} + 50°F or an RCS temperature of 200°F, whichever is larger. This method would yield 295°F as an acceptable temperature for 32 effective full-power years (EFPYs) of operation. Additionally, the licensee chose to set the LTOPS arming temperature conservatively to 350°F. Because these values are conservative and because they satisfy the 10 CFR Part 50, Appendix G requirements, the staff finds them acceptable.

3.1.2.2 Sequoyah Unit 2

The proposed PTLR is included as Enclosure 4 of Reference 1. The licensee based the update to these limits and setpoints on the Westinghouse topical report, WCAP-15321 (Reference 13). Both the P/T limits and the LTOP setpoints were calculated using the methodology described in Reference 8, Revision 2, as supplemented by: (1) the ASME Code Case N-640 “Alternate Reference Fracture Toughness for Development of PT Limits for Section XI, Division 1,” (2) the 1996 version of ASME Section XI, Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” and (3) WCAP-15315, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants” (Reference 11).

3.1.2.2.1 WCAP-15321, Revision 1, “Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation”

3.1.2.2.1.1 Appendix C, "Calculated Fluence Data"

The licensee developed the best estimate exposure for the SQN, Unit 2 reactor vessel in WCAP-15320 (Reference 14), using a combination of absolute plant specific transport calculations and plant specific measurement data.

The staff found the pressure vessel projected fluence values to be acceptable because the calculation was based on the DORT code, a 2-D discrete ordinates in the (r,θ) and (r, z) planes. Additionally: (1) the methodology is consistent with the guidance of Reference 9, (2) the code approximations (i.e., P3 for the anisotropic scattering expansion and S_8 for the angular quadrature) are acceptable, (3) the cross section file (BUGLE-96) was based on ENDF/B-VI, which is recommended in Reference 9, and (4) the results indicate that the M/C ratio for E>1.0 MeV is slightly greater than 1, which is conservative.

In summary, the staff reviewed the submittal to determine the applicability of the LTOP limits and the fluence calculations. The staff review determined that the fluence calculations are consistent with the guidance of RG 1.190, and therefore, the calculated values are acceptable. Based on this conclusion the staff also finds that the LTOP limits are valid for both SQN, Units 1 and 2.

3.1.2.2.2 Low Temperature Overpressure Protection System (LTOPS) Setpoints (TS 3/4.4.12)

The pressurizer PORV lift setpoints are determined from the P/T curves and satisfy 10 CFR 50, Appendix G. The setpoints do not include instrument uncertainties. However, TVA quantified the instrument channel uncertainties and evaluated the setpoints versus the limits of 10 CFR Part 50, Appendix G. The calculated enable temperature could be based on the ASME Code Case N-514 requirements. These would allow SQN, Unit 2 to use RT_{NDT} + 50°F or an RCS temperature of 200°F, whichever is larger. This method would yield 225°F as an acceptable temperature for 32 EFPYs of operation. Additionally, the licensee chose to set the LTOPS arming temperature conservatively to 350°F. Because these values are conservative and because they satisfy the Appendix G requirements, the staff finds them acceptable.

3.1.3 Summary

The staff reviewed the licensee's proposal to implement a PTLR for SQN, Units 1 and 2. Based on this review, the staff finds that the licensee's methodology for neutron fluence and LTOP system calculations, which is consistent with the guidance of RG 1.190 and WCAP-14040-P-A, satisfies the requirements of GL 96-03. Because the neutron fluence and LTOP system calculations were developed with staff approved methodologies, the staff finds them acceptable.

3.2 <u>Technical Specification Review</u>

3.2.1 Regulatory Evaluation

The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. This regulation requires that the TS include items in five specific categories. These categories include (1) safety limits, limiting safety system settings and limiting control settings, (2) limiting conditions for operation, (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls. Administrative controls are the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in 10 CFR 50.4.

The regulation does not specify the particular TSs to be included in a plant's license. In addition, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether an LCO is required to be included in the TS. These criteria are as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Existing LCOs and related surveillances included as TS requirements which satisfy any of the criteria stated above must be retained in the TSs. Those TS requirements which do not satisfy these criteria may be relocated to other, licensee-controlled documents.

Reference 7 allows licensees to relocate the P/T limit curves from their plant TSs to a PTLR or a similar document. The LTOP system limits were also allowed to be relocated to the same document. The methodology used to determine the P/T and LTOP system limit parameters must comply with the specific requirements of Appendices G and H to Part 50 of 10 CFR, be documented in an NRC-approved topical report or an NRC approved plant-specific submittal, and be incorporated by reference into the TSs. Subsequent changes in the methodology must be approved by a license amendment.

According to this guidance, the applicant must have obtained NRC review and written approval of the P/T methodology and the proposed PTLR before the NRC can approve TS changes associated with establishing a PTLR. The associated changes affect the definitions, LCOs, and administrative controls sections of the TS. Specifically, the applicant must modify its plant TSs by adding:

- In the definitions section, the definition of a named formal report (PTLR or a similar document) that would contain the explanations, figures, values, and parameters (currently contained in TSs) derived in accordance with an NRC-approved methodology and consistent with all of the design assumptions and stress limits for cyclic operation;

- In affected LCOs, references to the PTLR that require maintaining the P/T limits within the limits specified in the PTLR, in place of the existing P/T limits explanations, figures, values, and parameters; and

- In the administrative controls section, a reporting requirement to submit the PTLR to the NRC, when it is issued, for each reactor vessel fluence period. The PTLR administrative controls specification must reference the document from the NRC that approved the supporting P/T methodology.

3.2.2 Technical Evaluation

3.2.2.1 TS Changes Associated with the Establishment of a PTLR

The licensee proposed to relocate the current SQN RCS P/T limit curves and LTOP system limits to a PTLR in accordance with Reference 7. The licensee-proposed TS revisions are based on NUREG-1431 and include the following:

(1) The definitions section of the TSs is modified to include a definition of the PTLR to which the figures, values, and parameters for P/T and LTOP system limits will be relocated on a unit-specific basis. These figures, values, and parameters are established in accordance with an NRC-approved methodology that maintains the P/T acceptance limits and the P/T limits of the safety analysis. As noted in the definition, plant operation within these limits are addressed by individual specifications. For SQN, Units 1 and 2, the definition of the PTLR (TS 1.23) is as follows:

> The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the LTOP arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.1.15.

(2) The following SQN, Units 1 and 2 specifications are revised to replace the numerical values of the P/T and LTOP system limits with a reference to the PTLR that provides these values (the associated Bases are also modified). The following figures are also relocated to the PTLR:

LCO 3.4.9.1	RCS Pressure and Temperature Limits
Figure 3.4-2	RCS Heatup Limitations Applicable up to 16 EFPY
Figure 3.4-3	RCS Cooldown Limitations Applicable up to 16 EFPY
3/4 4.12	Low Temperature Over Pressure Protection (LTOP) System Applicability, Required Action b, and footnote 2.
Figure 3.4-4	PORV Nominal Lift Settings - Applicable up to 16 EFPY

(3) TS 6.9.1.15, “Reactor Coolant System (RCS) Pressure and Temperature Limits (PTLR)

Report," is added to the reporting requirements of the administrative controls section of the TSs. Per Reference 6, the licensee has proposed to revise TS 6.9.1.15 to allow the NRC-approved documents (topical reports) that contain analytical methods used to develop the PTLR to be listed by number and title in 6.9.1.15.a. This would allow the licensee to use current approved topical reports to support limits in the PTLR without having to submit an amendment to the facility operating license every time the Topical Report is revised. The PTLR would provide the specific information identifying the particular approved topical reports used to determine the P/T limits or LTOP system limits. This still provides assurance that only the approved versions of the referenced plant specific methodologies will be used for the determination of the P/T limits or LTOP system limits since the complete citation will be provided in the PTLR, and those limits must be approved by the NRC.

The PTLR provides the explanations, figures, values, and parameters of the P/T and LTOP system limits for the applicable effective fluence period. Furthermore, this specification requires the figures, values, and parameters to be (a) established using the SQN plant-specific methodology reviewed and approved by the NRC, and (b) consistent with all applicable acceptance limits and the limits of the SQN safety analyses. Finally, this specification requires the licensee to document in the PTLR all changes in the values of these limits each effective fluence period and submit to the NRC the revised PTLR within 30 days of issuance.

On this basis, the NRC staff concludes that the licensee has proposed, consistent with Reference 7, an acceptable means of maintaining the detailed values of the current P/T limit curves and LTOP system limits, and making changes to these limits, as needed, in the future. Therefore, moving the values of P/T limits and LTOP system limits to the PTLR will not impact plant safety.

The information discussed above relating to the P/T limits and LTOP system limits is not itself required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. The previously listed LCOs, which satisfy one or more of the four criteria in 10 CFR 50.36(c)(2)(ii), will remain in TSs. These LCOs, consistent with 10 CFR Part 50, Appendix G P/T requirements, will continue to require operating the plant in accordance with the PTLR P/T limits and LTOP system limits. These limits will be maintained and revised using the NRC-approved methodology, as required by TS 6.9.1.15, or NRC prior approval of a license amendment to revise P/T limits and methodology must be obtained. Accordingly, the staff concludes that the detailed values of the current P/T limit curves and LTOP system limits may be removed from TSs and maintained in the PTLR. Therefore, the proposed PTLR and associated TS changes are acceptable. Along with the above changes, the licensee also proposed appropriate changes to the TSs Table of Contents (index and figure index) and TS Bases, including relocating Bases, Tables, and Figures to the PTLR. These changes are administrative and are, therefore, acceptable.

The staff also concludes that the relocated requirements discussed above relating to the P/T limits and LTOP system limits are not required to be in the TSs under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Accordingly, the staff concludes that the proposed changes are acceptable and that these requirements may be relocated from the TSs to the PTLR.

3.2.2.2 Relocation of TS 3/4.4.9.2, Pressurizer

The existing TS 3/4.4.9.2 conditions, actions, and SRs for the pressurizer temperature limits will be relocated to the SQN TRM. These requirements define the temperature limitations on the pressurizer heatup and cooldown, and spray water temperature differential to assure that the pressurizer remains within the design criteria assumed for the pressurizer fatigue analysis performed in accordance with the ASME Code requirements.

The staff evaluated the existing TSs against the four criteria set forth in 10 CFR 50.36(c)(2)(ii). Pressurizer temperature limits are not a form of instrumentation nor a structure, system or component, and therefore, do not meet criteria 1, 3 or 4. The pressurizer temperature limits are process variables. However, these process variables are consistent with boundaries assumed in the structural analyses of the pressurizer and not as an initial condition for a design basis accident or transient analysis. Therefore, pressurizer temperature limits do not meet criterion 2 for inclusion in the TSs. Since TS 3/4.4.9.2 requirements do not satisfy these criteria, TS 3/4.4.9.2 may be relocated to the SQN TRM.

Changes to the TRM are controlled in accordance with approved station procedures and the requirements of 10 CFR 50.59. Therefore, the staff concludes that sufficient regulatory controls exists and concludes that TS 3/4.4.9.2 may be relocated from the TSs to the licensee's TRM.

3.2.2.3 Revisions to TS 3/4.4.9.1 and TS 3/4.4.12

The licensee proposed to revise, in its entirety, TS 3/4.4.9.1 and TS 3/4.4.12, to incorporate standard titles, LCO requirements, applicability, action requirements, surveillance requirements, and notation consistent with NUREG-1431. The NUREG-1431 format is not adopted with the proposed changes. The specific changes are as follows:

A. The current TS titles for 3/4.4.9.1, "Pressure/Temperature Limits," and 3/4.4.12, "Low Temperature Overpressure Protection Systems," are revised to incorporate the standard titles. Specifically, the titles will become "RCS Pressure and Temperature (P/T) Limits," and "Low Temperature Overpressure Protection (LTOP) System," respectively. These changes are administrative in nature, do not affect the existing TS requirements, and therefore, are acceptable.

B. The current LCO 3.4.9.1 is replaced with the standard LCO wording. LCO 3.4.9.1 currently states:

   "The Reactor Coolant system (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

   a. A maximum heatup of 100°F in any one hour period.

   b. A maximum cooldown of 100°F in any one hour period.

   c. A maximum temperature change of less than or equal to 5°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves."

As discussed in Section 3.2.2.1(2) above, TS 3.4.9.1 is revised to relocate the P/T limits and associated figures to the PTLR. The revised LCO will state: “RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.” Once the P/T limits and associated figures are relocated to the PTLR, the proposed LCO wording change becomes administrative in nature, does not affect the TS requirements, and therefore, is acceptable.

d. The current action in TS 3.4.9.1 is replace with two action statements. Action a will be associated with ensuring the limits are met while in MODES 1, 2, 3, and 4, and Action b is associated with any time other than MODES 1, 2, 3, and 4. The current action requirement states:

“With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{ave} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.”

The proposed action statements are consistent with NUREG-1431:

“a. With the requirements of the LCO not met in MODE 1, 2, 3, or 4, restore the parameter(s) to within limits in 30 minutes and determine RCS is acceptable* for continued operation within 72 hours. With the required action above not met, be in MODE 3 within the next 6 hours and in MODE 5, with RCS pressure < 500 psig, within the following 30 hours.

b. With the requirements of the LCO not met any time other than MODE 1, 2, 3, or 4, immediately initiate action to restore parameter(s) to within limits and, prior to entering MODE 4, determine RCS is acceptable* for continued operation.”

The requirements of proposed Action a are consistent with the current TS 3.4.9.1 action statement for MODES 1, 2, 3, and 4. In addition, the licensee proposed a 72-hour completion time for the determination that the RCS is acceptable for continued operation. The current TS does not have a completion time for the determination. The 72-hour completion time provides a reasonable amount of time to complete the determination, is a more restrictive change to the current TS, and is acceptable. As such, the current TS requirements are maintained and the proposed change in wording for Action a is also acceptable.

The requirements of proposed Action b are more restrictive than the current TS since proposed Action b does not allow a 30-minute completion time to restore the parameter(s) to within limits. Since proposed Action b is applicable to MODES 5 and 6, the shutdown track specified in the current TS action statement is not required. All other requirements proposed by Action b are consistent with the current TS action statement. Since the current TS requirements are maintained or made more restrictive, the staff concludes that the proposed wording for Action b is acceptable.

Both Actions a and b include a footnote *. The proposed footnote will state "The determination that the RCS is acceptable for continued operation must be completed for any entry into Action a or b." Although the format is not the same, the footnote is consistent with NUREG-1431. Since the current TS only has one Action statement, the footnote was not required. With two Action statements, the footnote emphasizes the importance of the determination for continued operation and, therefore, is acceptable.

c. The current surveillance requirement, SR 4.4.9.1.1, is revised to incorporate the NUREG-1431 wording. SR 4.4.9.1.1 currently states:

   "The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations."

   The revised SR 4.4.9.1.1 will read: "Verify** RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR every 30 minutes." The proposed footnote ** will state "Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing." Although the format is not the same, the footnote is consistent with NUREG-1431. With the proposed footnote, the proposed revision to the SR maintains the current SR 4.4.9.1.1 requirements and, therefore, is acceptable.

d. The current SR 4.4.9.1.2 states: "The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine change in material properties, in accordance with 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3, and 3.4-4." The licensee proposed to delete SR 4.4.9.1.2 because it duplicates the programmatic requirements within the SQN TSs and 10 CFR Part 50, Appendix H. In addition, the proposed PTLR will contain Figures 3.4-2, 3.4-3, and 3.4-4 and the reactor vessel surveillance capsule withdrawal schedules for each unit. On this basis, the staff concludes that the deletion of SR 4.4.9.1.2 is acceptable and is consistent with NUREG-1431.

e. The current LCO 3.4.12 is replaced with the standard LCO wording. LCO 3.4.12 currently states:

"At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a nominal lift setting less than or equal to that shown in Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 3 square inches."

As discussed in Section 3.1(2) above, TS 3.4.12 is revised to relocate the LTOP system limits and associated figure to the PTLR. The revised LCO will state:

"3.4.12* An LTOP System shall be OPERABLE with a maximum of one centrifugal charging pump and no safety injection pump capable of injecting into the Reactor

Coolant System (RCS) and the accumulators isolated and one of the following pressure relief capabilities:

a. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

b. The RCS depressurized and an RCS vent ≥ 3 square inches."

Once the LTOP system limits and associated figure are relocated to the PTLR, the proposed LCO 3.4.12.a wording change becomes administrative in nature, does not affect the TS requirements, and therefore, is acceptable. The LCO statement that a maximum of one centrifugal charging pump and no safety injection pump shall be capable of injecting into the RCS for the LTOP system to be OPERABLE is consistent with the current TS 3.4.12 Action e which states: "When RCS temperature is less than 350°F, both safety injection pumps and one centrifugal charging pump shall be made incapable of automatic injection into the RCS." Therefore, this change is consistent with the current licensing basis. In addition, the proposed LCO incorporates a requirement for isolation of the cold leg accumulators. This is a more restrictive change that increases the protection against mass injection from these components and therefore is acceptable.

The LCO statement includes a footnote *. The proposed footnote will state:

"1) Two charging pumps may be made capable of injecting into the RCS for ≤ 1 hour for pump swap operations.

2) Accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

3) For the purpose of making the required safety injection pumps and charging pump inoperable, the following time is permitted: up to 4 hours after entering MODE 4 from MODE 3, or prior to decreasing temperature on any RCS loop to below 325°F, whichever occurs first."

Although the format is not the same, Footnotes 1) and 2) are consistent with NUREG-1431. Current TS 3.4.12 Action e allows 12 hours to make any of the safety injection or charging pumps, that are capable of injection, incapable of injection when the RCS temperature is less than 350°F. Footnote 1) provides an allowance for charging pump swap operations. Pump swap operations may be necessary to maintain continuous charging to the RCS and continuous reactor coolant pump (RCP) seal flow, which is required to be maintained at all times. When a charging pump swap is necessary, procedures require starting a second charging pump and allowing it to stabilize before stopping the first pump. This process ensures continuous RCP seal flow while administratively controlling charging pump alignment to prevent inadvertent mass injection to the RCS. The 1-hour time period provides sufficient time to safely complete the transfer and to complete administrative controls and surveillances associated with the swap. Footnote 2) provides operational flexibility. Administrative controls are available to the operator for RCS pressure control to minimize the risk of injecting an unisolated cold accumulator during the lower modes of plant operation. Based on the above, the staff concludes that proposed Footnotes 1) and 2) are acceptable.

Footnote 3) is based on the SQN LTOP system design and the arming temperature contained in the PTLR. A similar Note is included in STS 3.5.2, “ECCS [Emergency Core Cooling System] - Operating” which requires that two independent ECCS trains be operable in MODES 1, 2, and 3. The STS 3.5.2 Note allows operation in MODE 3 with ECCS trains made incapable of injecting in order to facilitate entry into or exit from the Applicability of LCO 3.4.12. When the LTOP arming temperature is at or near the MODE 3 boundary temperature of 350°F, time is needed to make pumps incapable of injecting prior to entering the LTOP Applicability, and to provide time to restore the inoperable pumps to OPERABLE status on exiting the LTOP Applicability.

The proposed TS 3.4.12 Action a requires the licensee to immediately take action to make two safety injection pumps and at least one centrifugal charging pump be incapable of injection when in MODE 4 with the cold leg temperature less than or equal to the LTOP arming temperature. SQN’s LTOP system is required to be armed at the RCS temperature of 350°F which is also the transition temperature between MODE 4 and MODE 3. Similar to the justification for the note in STS 3.5.2, during plant cooldown for transition into MODE 4 from MODE 3, a period of time is needed to comply with provisions of the LCO that require rendering the safety injection pumps and one centrifugal charging pump incapable of RCS injection. The addition of the Footnote 3) provides for 4 hours to render the safety injection pumps and one centrifugal charging pump to be incapable of RCS injection during a plant cooldown. In addition, the licensee proposed a temperature limit of 325°F to prevent continued cooling with operable pumps.

Current TS 3.4.12 Action e allows 12 hours to make any of the safety injection or charging pumps, that are capable of injection, incapable of injection when the RCS temperature is less than 350°F. During this time, the licensee would be allowed to continue RCS cooldown. The proposed Footnote 3) only allows 4 hours and limits the cooldown to 325°F. This 4-hour period is more conservative than the current TS 3.4.12 Action e and is adequate given the low probability for an inadvertent injection during that period. The 325°F cooldown limit is more conservative than the current TS 3.4.12 Action e requirements, which had no limits.

Based on the above discussion and the fact that the proposed Footnote 3) is more conservative than the current TS 3.4.12 Action e, the staff concludes that Footnote 3) is acceptable.

c. The current TS 3.4.12 Applicability is revised to adopt the NUREG-1431 wording which changes the applicability in MODE 4 from “MODE 4" to “MODE 4 when any RCS cold leg temperature is ≤ the LTOP arming temperature specified in the PTLR.” The proposed change is appropriate since it aligns LTOP applicability to the LTOP design. The LTOP arming temperature will be maintained in PTLR which is consistent with GL 96-03. Therefore, the staff concludes that the proposed change is acceptable.

d. Most of the current TS 3.4.12 Action statements are replaced with the wording of the

Action statements in NUREG-1431. The proposed changes are as follows:

Current Action a requires: “With one PORV inoperable, in MODE 4 either: 1. Restore the inoperable PORV to operable status within 7 days or 2. Depressurize and vent the RCS through at least 3 square inches within the next 8 hours, or 3. Ensure pressurizer level is maintained less than or equal to 30 percent.” Current Action a.1 will become Action c which states: “With one required PORV inoperable in MODE 4, restore the required PORV to OPERABLE status within 7 days.” The option to ensure pressurizer level is maintained less than or equal to 30 percent is not maintained in the revised TS 3.4.12. In addition, Action a.2 is captured in the revised Action e discussed below.

Current Action b requires: “With one PORV inoperable in MODES 5 or 6, either (1) restore the PORV to operable status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 3 square inch vent within a total of 32 hours. Current Action b.1 will become Action d which states: ”With one required PORV inoperable in MODE 5 or 6, restore the required PORV to OPERABLE status within 24 hours.” Current Action b.2 is captured in the revised Action e discussed below.

Current Action c requires: “With both PORVs inoperable, depressurize and vent the RCS through at least a 3 square inch vent within 8 hours.” Current Action c will be incorporated into revised Action e which states: “With two required PORVs inoperable, or the Actions (a), (b), (c), or (d) not met, or the LTOP System inoperable for any reason other than (a), (b), (c), or (d), depressurize the RCS and establish RCS vent of $\geq$ 3.0 square inches within 12 hours.”

Current Action e requires: “When RCS temperature is less than 350°F, both safety injection pumps and one centrifugal charging pump shall be made incapable of automatic injection into the RCS. Should any of these pumps be found actually capable of automatic injection, return the pump(s) to incapable status within 12 hours or depressurize and vent RCS through at least a 3 square inch vent within the next 8 hours.” Current Action e will become revised Action a which states: “Should one or more safety injection pumps or more than one charging pump be found capable of injecting into the RCS, immediately initiate action to verify a maximum of one centrifugal charging pump and no safety injection pumps are capable of injecting into the RCS.” The second part of current Action e is captured in revised Action e discussed below.

The staff has reviewed the proposed wording changes to current TS 3.4.12 Actions a, b, c, and e to adopt the NUREG-1431 wording. The proposed wording changes are administrative in nature, does not affect the TS requirements, and therefore, is acceptable.

e. The completion time to establish an RCS vent of $\geq$ 3.0 square inches has been extended from the current completion time of 8 hours in Actions a, b, c, and e to 12 hours in revised Action e. This incremental change in completion time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements and, therefore, is acceptable.

f. Current Action d requires: “With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise verify the vent path every 12 hours.” Current Action d will not be retained as an Action Statement but will be captured in a new SR, SR 4.4.12.5, which states: “Verify# required RCS vent $\geq$ 3.0 square inches open at least: a. Once every 12 hours for unlocked open vent valve(s) and, b. Once every 31 days for other vent path(s).” The staff finds the new SR 4.4.12.5 to be appropriate since it properly reflects the requirements of LCO 3.4.12, and therefore is acceptable.

SR 4.4.12.5 is modified by a footnote, #, which states: “Only required to be met when complying with LCO 3.4.12.b.” The proposed footnote provides additional information as to when the SR is required to be performed and is acceptable. The proposed SR 4.4.12.5 is also consistent with NUREG-1431.

g. Current Action f requires: “In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.” The licensee proposed to delete current Action f because this information is currently reported in Licensee Event Reports and would be a duplicate requirement. In addition, the removal of this requirement is based on NRC approved TSTF-285R4 and GL 97-02 which identify information that is required to be reported to the NRC. Therefore, the staff finds the deletion of current Action f to be acceptable.

h. Current Action g requires: “The provisions of Specification 3.0.4 are not applicable.” This action statement is retained in the revised TS 3.4.12 and will become revised Action f.

i. Current SR 4.4.12.1 is maintained in the revised TS 3.4.12 with minor wording changes to adopt the NUREG-1431 wording. The revised SR 4.4.12.1 will state:

“Each PORV shall be demonstrated OPERABLE by: a. Performance of a CHANNEL FUNCTIONAL TEST,* but excluding valve operation, at least once per 31 days; b. Performance of a CHANNEL CALIBRATION on each required PORV actuation channel at least once per 18 months, and c. Verifying the PORV block valve is open for each required PORV at least once per 72 hours.”

The CHANNEL FUNCTIONAL TEST specified in SR 4.4.12.1.a is consistent with the current SR 4.4.12.1 and therefore is acceptable. The licensee has proposed a footnote, *, which states: “Not required to be performed until 12 hours after decreasing RCS cold leg temperature to $\leq$ the LTOP arming temperature in the PTLR.” Although not specifically addressed by the licensee, the proposed footnote accounts for the fact that the CHANNEL FUNCTION TEST cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.” Based on the above, the staff concludes that the footnote is acceptable for SR 4.4.12.1.a. The footnote is also consistent with NUREG-1431 wording.

j. Current SR 4.4.12.2 is replaced in its entirety. SR 4.4.12.2 currently requires: "All charging pumps and safety injection pumps, excluding the required OPERABLE pumps per specification 3.1.2.3 and 3.5.3, shall be verified incapable of injecting into the RCS and the cold leg accumulator discharge valves verified closed and locked out at least once per 31 days except when the reactor vessel head is removed by verifying that either the pump controls are in the pull-to-lock position, the pump motor circuit breaker(s) is tagged out or the pump(s) is isolated from the RCS by a manually closed valve or by a motor-operated valve with the valve breaker tagged out. Normal Reactor Coolant Pump seal flow can be maintained at all times."

   In its place, the licensee has proposed two SRs which will state the following:

   SR 4.4.12.2: "Verify no safety injection pumps are capable of injecting into the RCS within 4 hours after entering MODE 4 from MODE 3 and prior to the temperature of one or more RCS cold legs decreasing below 325°F, and every 12 hours thereafter."

   SR 4.4.12.3: "Verify a maximum of one charging pump is capable of injecting into the RCS within 4 hours after entering MODE 4 from MODE 3 and prior to the temperature of one or more RCS cold legs decreasing below 325°F, and every 12 hours thereafter."

   The proposed SRs verify ECCS pumps are rendered incapable of injecting into the RCS consistent with the LCO requirement. This verification is required to minimize the potential for an LTOP event by limiting the mass input capability. The proposed frequency of within 4 hours after entering MODE 4 from MODE 3 and prior to the temperature of one or more RCS cold legs decreasing below 325°F is consistent with Footnote 3) to the LCO and is sufficient to verify the required status of the equipment. The proposed SRs 4.4.12.2 and 4.4.12.3 are sufficient to demonstrate that the requirements of LCO 3.4.12 are being met and, therefore, are acceptable.

k. The licensee has proposed to add a new SR, SR 4.4.12.4 which requires: "Verify each accumulator is isolated at least once per 12 hours." Although a similar requirement was captured in current SR 4.4.12.2, the new SR 4.4.12.4 specifically addresses the isolation of the accumulators which is part of the revised LCO 3.4.12. The proposed SR 4.4.12.4 is sufficient to demonstrate that the requirements of LCO 3.4.12 are being met and, therefore, is acceptable.

3.2.3 Summary

Based on the above discussion, the NRC staff concludes that the licensees proposed changes to the TSs are acceptable. These changes include (1) the relocation of the P/T limit curves and LTOP system limits to the PTLR, (2) the referencing of the PTLR in the affected TS LCOs and bases, including the addition of the PTLR to the definitions section of the TSs, and the addition of a new TS 6.9.1.15 to the administrative controls section of the TS, (3) the relocation of TS 3/4.4.9.2, Pressurizer, to the SQN TRM, and (4) the revision of TS 3/4.4.9.1, Pressure/Temperature Limits, Reactor Coolant System, and TS 3/4.4.12, Low Temperature Over Pressure Protection Systems, to incorporate standard TS requirements from NUREG-1431, Revision 2.

3.3 Material Engineering Evaluation

3.3.1 Regulatory Evaluation

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the acceptability of a facility's proposed PTLR methodology and initial PTLR based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Appendix H to 10 CFR Part 50; RG 1.99, Revision 2 (RG 1.99, Rev. 2); Generic Letter (GL) 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Standard Review Plan (SRP) Section 5.3.2; and GL 96-03. Appendix G to 10 CFR Part 50 requires that facility P/T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. GL 92-01, Revision 1, requested that licensees submit their RPV data for their plants to the staff for review, and GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. SRP Section 5.3.2 provides an acceptable method of determining the P/T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code.

Reference 7 addresses the technical information necessary for a licensee's implementation of a PTLR. Reference 7 establishes the information which must be included in: (1) an acceptable PTLR methodology (which will be used to develop the PTLR) and, (2) the information which must be included with the PTLR itself. These information requirements are principally addressed in a table contained in Reference 7, Attachment 1, entitled "Requirements for Methodology and PTLR," and are subdivided into seven Technical Elements which are numbered by rows in the table.

Reference 7 also addresses the appropriate modifications to the administrative controls section of a facility's TS which are necessary to implement a PTLR. TSTF-419 provides guidance on an alternative set of facility TS administrative control section changes which may be made to implement a PTLR. Review of the proposed modifications to the administrative controls section of the facility's TS is provided by the NRC's Technical Specification Section.

Per References 1, 5 and 6, TVA provided the following information which was reviewed by the staff:

(1) A license amendment request including proposed TS 6.9.1.15 for SQN, Unit 1 and 2 which identifies the documents which fully describe the PTLR methodology for the units, and

(2) The initial versions of the proposed SQN, Unit 1 and 2 PTLRs which indicate the results obtained from the licensee's proposed PTLR methodology.

The most recent initial versions of the proposed SQN, Unit 1 and Unit 2 PTLRs were submitted as attachments to Reference 5.

Regarding item (2), the most recent initial versions of the proposed SQN, Unit 1 and Unit 2 PTLRs ("Tennessee Valley Authority, Sequoyah Unit 1, Pressure Temperature Limits Report, Revision 4, July 2003," and "Tennessee Valley Authority, Sequoyah Unit 2, Pressure Temperature Limits Report, Revision 5, July 2003,") were submitted as attachments to Reference 5. These revisions of the proposed PTLRs were reviewed by the NRC staff against the criteria in Reference 7.

The licensee concluded in its submittal, as revised by References 5 and 6, that the information provided was sufficient to address the regulatory requirements for the implementation of a PTLR.

3.3.2 Technical Evaluation

3.2.2.1 Reactor Vessel Material Surveillance Program

Concerning the licensee's reactor vessel material surveillance program, Reference 7 states that, at a minimum, a licensee's PTLR methodology shall:

> Briefly describe the RPV [reactor pressure vessel] surveillance program. The licensee should identify by title and number the report containing the RPV surveillance program and surveillance capsule reports.

The NRC staff concluded in its most recent safety evaluation on WCAP-14040 (dated February 27, 2004, approving Revision 3 of the topical report) that:

> The provisions of the methodology described in WCAP-14040, Revision 3, do not specify how the plant-specific RPV surveillance programs should be maintained in order to be in compliance with Appendix H to 10 CFR Part 50. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must submit additional information to address the methodology requirements discussed in provision 2 in the table of Attachment 1 to GL 96-03 related to the RPV material surveillance program.

TVA, however, included additional plant-specific information regarding the SQN, Unit 1 and 2 RPV material surveillance programs in References 10 and 13, respectively. The information provided by the licensee in References 10 and 13 adequately addresses the reactor vessel material surveillance program technical element specified in GL 96-03. Hence, since TVA will include References 10 and 13 in its PTLR methodology, the staff concludes that this criteria is satisfied.

Reference 7 also states that, at a minimum, a licensee's PTLR shall:

> Provide the surveillance capsule withdrawal schedule, or reference by title and number the documents in which the schedule is located. Reference the surveillance capsule reports by title and number if RPV material adjusted reference temperatures (ARTs) are calculated using surveillance data.

The NRC reviewed the information provided in the draft SQN, Unit 1 and 2 PTLRs. TVA referenced all applicable surveillance capsule reports which provide information relevant to the calculation of SQN, Unit 1 and 2 RPV material ARTs in Section 8.0 of the draft PTLRs. Hence, the staff concludes that this criteria is satisfied.

3.3.2.2 Calculation of RPV Material ARTs

Concerning the licensee's calculation of RPV material ARTs, Reference 7 states that, at a minimum, a licensee's PTLR methodology shall:

> Describe the method for calculating the ARTs using NRC RG 1.99, Revision 2.

The NRC staff concluded in its February 27, 2004, safety evaluation (SE) concerning WCAP-14040, Revision 3, that Reference 8 was adequate to meet the minimum requirements for a licensee's PTLR methodology for this technical element. Hence, since TVA will include Reference 8 in its PTLR methodology, the staff concludes that this criteria is satisfied.

Reference 7 also states that, at a minimum, a licensee's PTLR shall:

> Identify both the limiting ART values and limiting materials at the 1/4t and 3/4t locations (i.e., locations 1/4 of the way through the thickness of the ferritic RPV wall from the inside and outside surface) and PWRs shall identify the RPV's limiting RT_{PTS} value in accordance with 10 CFR 50.61.

The required information was provided in Section 4.0 of the draft PTLRs. Hence, the staff concludes that this criteria is satisfied.

3.3.2.3 Calculation of P/T Limit Curves Based on Limiting Material ART values

Concerning the licensee's calculation of P/T limit curves based on limiting material ART values, Reference 7 states that, at a minimum, a licensee's PTLR methodology shall:

> Describe the application of fracture mechanics in constructing P/T limit curves based on Appendix G to Section XI of the ASME Code and NRC SRP Section 5.3.2.

In the NRC staff's February 27, 2004, SE (regarding WCAP-14040, Revision 3) it was stated that:

> [T]he NRC staff has concluded that the basic methodology specified in WCAP-14040, Revision 3, for establishing P/T limit curves meets the regulatory requirements of Appendix G to 10 CFR Part 50 and the guidance provided in SRP Section 5.3.2. However, the NRC staff has concluded that the discussion provided in WCAP-14040, Revision 3, regarding the use of optional guidelines for the development of P/T limit curves, including the use of ASME Code Cases N-588, N-640, and N-641 is not acceptable. The NRC staff has concluded, based on guidance provided by the NRC's Office of the General Counsel, that licensees do not need to obtain exemptions to use the provisions of ASME Code Case N-588, N-640, or N-641. The basis for this decision is as follows. Appendix G to 10 CFR Part 50 references the use of ASME Code Section XI, Appendix G and defines the acceptable Editions and Addenda of the Code by

> reference to those endorsed in 10 CFR 50.55a. The 2003 Edition of 10 CFR Part 50, 10 CFR 50.55a, endorses editions and addenda of ASME Section XI up through the 1998 Edition and 2000 Addenda. The provisions of N-588, N-640, and N-641 have been directly incorporated into the Code in the 2000 Addenda version of ASME Section XI, Appendix G. Therefore, licensees may freely make use of the provisions in Code Cases N-588, N-640, and N-641 by using the methodology in the 2000 Addenda version of ASME Section XI without the need for an exemption. When published, the approved revision of Topical Report WCAP-14040 should be modified to reflect this NRC staff conclusion.

The correction cited by the staff regarding WCAP-14040, Revision 3 does not affect the technical adequacy of the methodology specified in the topical report. Hence, since TVA will include Reference 8 in its PTLR methodology, the staff concludes that this criteria is satisfied.

Reference 7 also states that, at a minimum, a licensee’s PTLR shall:

> Provide the P/T limit curves for heatup, cooldown, criticality, and hydrostatic and leak rate testing.

In Section 5.0 of the licensee’s draft PTLRs, Figures 2-1 and 2-2 provide P/T limit curves for inservice hydrostatic testing and a P/T limit curve applicable to both heatup and cooldown of the RPV with the core not critical which were developed using the licensee’s proposed PTLR methodology. These P/T limit curves were reviewed and found to be acceptable by the NRC staff in that they comply with the requirements specified in 10 CFR Part 50, Appendix G. Hence, the staff concludes that this criteria is satisfied.

3.3.2.4 P/T Limit Curve Minimum Temperature Requirements

Concerning the licensee’s incorporation of P/T limit curve minimum temperature requirements as specified by Appendix G to 10 CFR Part 50, Reference 7 states that, at a minimum, a licensee’s PTLR methodology shall:

> Describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P/T limit curves.

The NRC staff concluded in its February 27, 2004, SE concerning Reference 8 that WCAP-14040, Revision 3 was adequate to meet the minimum requirements for licensee’s PTLR methodology for this technical element. However, per Reference 2, TVA requested that SQN, Units 1 and 2 be exempted from the minimum temperature requirements related to RPV closure flange region material properties (see Footnote 2 to Table 1 in 10 CFR Part 50, Appendix G) based on the technical information provided in Reference 15. By letter dated July 7, 2004, the NRC staff approved the TVA exemption request for SQN, Units 1 and 2 and modified the applicability of the minimum temperature requirements in Appendix G to 10 CFR Part 50 based on the information provided in Reference 15. Therefore, the NRC staff concludes that the minimum temperature requirements specified in Reference 8, as modified by Reference 15, meet the minimum requirements for a licensee’s PTLR methodology for this technical element. Hence, since TVA will include References 8 and 15 in its PTLR methodology, the staff concludes that this criteria is satisfied.

Reference 7 also states that, at a minimum, a licensee's PTLR shall:

> Identify minimum temperatures on the P/T limit curves such as minimum boltup temperature and hydrotest temperature.

In Section 5.0 of the licensee's draft PTLRs, Figures 2-1 and 2-2 provide P/T limit curves which include minimum temperature requirements specified in 10 CFR Part 50, Appendix G, as modified by the TVA exemption request discussed above. The staff reviewed the minimum temperature requirements incorporated into the SQN, Unit 1 and 2 PTLR Figures 2-1 and 2-2 and found to be acceptable in that they comply with the requirements specified in 10 CFR Part 50, Appendix G, as exempted. Hence, the staff concludes that this criteria is satisfied.

3.3.2.5 Evaluation and Use of RPV Surveillance Data

Concerning the licensee's evaluation and use of RPV surveillance data, Reference 7 states that, at a minimum, a licensee's PTLR methodology shall:

> Describe how the data from multiple surveillance capsules are used in the ART calculation. Describe the procedure used if measured values of transition temperature shift from the surveillance capsules exceed predicted values. If data from other facilities is being used, identify the facilities which are providing data and identify by title and number the NRC SE which approved the use of the data for the facility.

In the NRC staff's February 27, 2004, SE (regarding WCAP-14040, Revision 3) it was stated that:

> Requirement 2 of Section 2.4 of WCAP-14040, Revision 3, addresses the determination of changes in material properties due to irradiation. This information includes a description of how surveillance capsule test results may be used to calculate RPV material properties in a manner which is consistent with Section C.2.1 of RG 1.99, Revision 2, and other NRC staff guidance.
>
> The NRC staff has reviewed the information in Section 2.4 of the TR [topical report] and determined that it is consistent with NRC staff guidance, including RG 1.99, Revision 2, and is, therefore, acceptable.

Hence, based on TVA's incorporation of Reference 8 into the proposed SQN, Units 1 and 2 PTLR methodology, the staff concludes that this criteria is satisfied.

Reference 7 also states that, at a minimum, a licensee's PTLR shall:

> Provide supplemental data and calculations of the chemistry factor in the PTLR if the RPV surveillance data are used to establish RPV material ART values. The PTLR shall also include an evaluation of RPV surveillance data to determine if they meet the credibility criteria in RG 1.99, Revision 2 and the results of this evaluation.

In Section 5.0 of the licensee's draft PTLRs, Table 5-2 provides an evaluation of the RPV surveillance data relevant to SQN, Unit 1 and 2 to determine if the data meets the credibility criteria in RG 1.99, Revision 2 and the results of this evaluation. The staff reviewed the information provided in Table 5-2 of the SQN, Unit 1 and 2 PTLRs and found that it accurately reflected an assessment of the SQN, Unit 1 and 2 RPV surveillance data which was consistent with RG 1.99, Revision 2. Hence, the staff concludes that this criteria is satisfied.

3.3.3 Summary

Based on the NRC staff's review of the information provide in TVA's September 6, 2002, September 3, 2003, and October 22, 2003, submittals, the staff concludes that regarding review topics assessed in this SE:

(a) TVA has defined an acceptable PTLR methodology which is consistent with the regulatory requirements given in Section 3.3.2 of this SE. This acceptable methodology is documented in References 8,10,13, and 15 in section 3.3.2 of this SE.

(b) TVA provided as an attachment to its October 22, 2003, letter proposed PTLRs for SQN, Unit 1 and 2 which contain information consistent with NRC regulatory requirements and are acceptable for incorporation into the SQN, Unit 1 and 2 licensing bases.

Therefore, TVA should be permitted to implement their proposed PTLRs provided that appropriate documentation of the licensee's PTLR methodology is incorporated into the Administrative Control Section of the SQN, Unit 1 and 2 TSs.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official 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 the 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 amendments involve 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 (67 FR 66015; October 29, 2002). Accordingly, the amendments meet 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 reviewed the proposed amendment to TS 3/4 9.2, TS 3/4.10.3 and TS 3/4.10.4 at SQN and has found them acceptable. The Commission has concluded, based on the nature of the proposed changes, that: (1) there is a reasonable assurance that the health and safety of the public will no be endangered by operation in the proposed manner, (2) such activities will be concluded 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. Smith, J. D., TVA to NRC, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Technical Specification (TS) Change No. 00-14, "Pressure Temperature Limits Report (PTLR) and Request for Exemption from the Requirements of 10 CFR 50, Appendix G," September 6, 2002.

2. Salas, P., TVA to NRC, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Westinghouse Electric Company Topical Report (WCAP-15984) for Technical Specification (TS) Change No. 00-14, "Pressure Temperature Limits Report (PTLR) and Request for Exemption from the Requirements of 10 CFR 50, Appendix G," December 19, 2002.

3. Salas, P., TVA to NRC, "Sequoyah Nuclear Plant (SQN) - Response to Request for Additional Information (RAI) Regarding Technical Specification (TS) Change No. 00-14, Pressure Temperature Limits Report (PTLR) and Request for Exemption from the Requirements of 10 CFR 50, Appendix G," March 28, 2003.

4. Salas, P., TVA to NRC, "Sequoyah Nuclear Plant (SQN) - Revised Responses to NRC Questions 1, 2, and 3 For Technical Specification (TS) Change No. 00-14, "Pressure Temperature Limits Report (PTLR) and Revised Topical Report (WCAP-15984, Revision 1), Reactor Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2," June 24, 2003.

5. Salas, P., TVA to NRC, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Updated Pressure Temperature Limits Reports (PTLRs) an Topical Reports for SQN Technical Specification (TS) Change No. 00-14," September 3, 2003.

6. Salas, P., TVA to NRC, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Technical Specification (TS) Change No. 00-14, Supplemental Changes (TAC Nos. MB6436 and MB6437)," October 22, 2003.

7. GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.

8. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," June 1994.

9. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

10. WCAP-15293, Revision 1, “Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation,” April 2001.

11. WCAP-15315, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants,” October 1999.

12. WCAP-15224, “Analysis of Capsule Y from the Tennessee Valley Authority Sequoyah Unit 1 Reactor Vessel Radiation Surveillance Program,” June 1999.

13. WCAP-15321, Revision 1, “Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation,” April 2001.

14. WCAP-15320, ”Analysis of Capsule Y from the Tennessee Valley Authority Sequoyah Unit 2 Reactor Vessel Radiation Surveillance Program,” December 1999.

15. WCAP-15984, ”Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2.”

Principal Contributors: Kerri Kavanaugh
Matt Mitchell
Sean Peters

Date: September 15, 2004