July 31, 2003

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

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT REGARDING REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN CURVES (TAC NO. MB9484) (TS 03-08)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 277 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Unit 2. This amendment is in response to your application dated June 5, 2003.

This amendment revises the reactor coolant system heatup and cooldown curves (pressure-temperature (P-T) limits). The revision replaces the P-T limits that were analyzed for 14.5 Effective Full Power Years (EFPYs) with new limits analyzed for 32 EFPYs. In addition, the amendment includes corresponding changes to the Technical Specification (TS) figure associated with the Low Temperature Over Pressure Protection and the TS Bases.

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/

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

Docket No. 50-328

Enclosures: 1. Amendment No. 277 to License No. DPR-79
2. Safety Evaluation

cc w/enclosures: See next page

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.277
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 June 5, 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. 277, 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 no later than 15 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by K. Jabbour Acting for/

Allen G. Howe, 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: July 31, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 277

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

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

REMOVE	INSERT
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-35	3/4 4-35
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 4-11	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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 277 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-328

1.0 INTRODUCTION

By application dated June 5, 2003, the Tennessee Valley Authority (the licensee, TVA) proposed an amendment to the Technical Specifications (TSs) for Sequoyah Nuclear Plant, Unit 2 (SQN2). The requested changes would revise the reactor coolant system (RCS) heatup and cooldown curves (pressure-temperature (P-T) limits). The revision replaces the P-T limits that are currently analyzed for 14.5 Effective Full Power Years (EFPYs) with new limits analyzed for 32 EFPYs. In addition, the amendment includes corresponding changes to the TS figure associated with the Low Temperature Over Pressure Protection and the TS Bases.

The proposed changes to the P-T limits were based, in part, on the use of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME B&PV Code, Section XI, Code Case N-640," to modify requirements established in Appendix G to Section XI of the 1995 Edition through 1996 Addenda of the ASME B&PV Code. In a separate application, dated September 6, 2002, TVA requested an exemption from the requirements of Appendix G to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 in order to utilize ASME Code Case N-640. The staff granted this exemption on July 30, 2003.

2.0 REGULATORY EVALUATION

2.1 P-T LIMITS

The U. S. Nuclear Regulatory Commission (NRC) has established requirements in Appendix G to 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff consulted the following regulatory guidance during the evaluation of P-T limit curves: Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"; GL 92-01, Revision 1, "Reactor Vessel Structural Integrity "; GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity "; Regulatory Guide 1.99, Revision 2 (RG 1.99, Rev. 2), "Radiation Embrittlement of Reactor Vessel Materials"; and Standard Review Plan (SRP) Section 5.3.2,

"Pressure-Temperature Limits and Pressurized Thermal Shock." GL 88-11 advised licensees that the NRC staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in ductile-to-brittle transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data have been recorded in the NRC staff's Reactor Vessel Integrity Database which is used by the NRC staff for the review of P-T limit curves. Appendix G to 10 CFR Part 50 also incorporates, by reference, the requirements found in Appendix G to Section XI of the ASME B&PV Code as the basis for the establishment of facility P-T limit curves. TVA has adopted Editions and Addenda of the ASME B&PV Code up to the 1995 Edition and 1996 Addenda as part of the SQN2 licensing basis.

2.2 Low-Temperature Overpressure (LTOP) LIMITS

The regulatory requirements for fluence calculations are in General Design Criteria (GDC) 14, 30, and 31 of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." In March 2001, the staff issued RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Methodologies which adhere to the guidance in RG 1.190 satisfy the requirements of the GDC 14, 30, and 31. Fluence calculations are acceptable if they are done with approved methodologies or with methods which are shown to conform to the guidance in RG 1.190. An approved methodology for the calculation of the P-T curves and the LTOP limits is presented in Westinghouse Commercial Atomic Power Report No. WCAP-14040NP-A.

3.0 TECHNICAL EVALUATION

3.1 P-T LIMITS

The licensee's method for establishing the proposed SQN2 P-T limit curves was based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME B&PV Code, as modified by ASME B&PV Code Case N-640. The basic parameter of this methodology is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. Appendix G to Section XI of the ASME B&PV Code requires the postulation of a sharp surface flaw, normal to the direction of the maximum stress, having a depth equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. Therefore, the critical locations in the RPV beltline region for calculating P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. Appendix G requires a structural factor of 2.0 on stress intensities resulting from reactor pressure (K_{Im}) during normal and transient operating conditions. A structural factor of 1.5 on stress intensities resulting from reactor pressure is applied for hydrostatic/leak test conditions. A structural faction of 1.0 is applied on stress intensities resulting from thermal loads (K_{It}) under either normal and transient operating or hydrostatic/leak test conditions. TVA's application included information concerning throughwall temperature gradients resulting from heatup and cooldown transients and the determination of the applied stress intensity at the tip of the postulated 1/4T and 3/4T flaws due to thermal loading (K_{It}).

The licensee requested, pursuant to 10 CFR 50.60(b), an exemption to use ASME Code Case N-640 as the basis for establishing the SQN2 P-T limit curves. ASME Code Case N-640 permits application of the lower bound static initiation fracture toughness (K_{IC}) curve as the basis for establishing the P-T curves in lieu of using the lower bound crack arrest fracture toughness (K_{IA}) curve which is invoked by 1995 Edition through 1996 Addenda of Appendix G to Section XI of the ASME Code.

Use of the K_{IC} curve in determining the lower bound fracture toughness curve in the development of P-T operating limits is more technically correct than use of the K_{IA} curve. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of an RPV. The staff concluded in the exemption dated July 30, 2003, that P-T curves based on the K_{IC} fracture toughness curve referenced by ASME B&PV Code Case N-640 are adequate to protect the RPV from brittle failure, and that the use of the K_{IC} curve, when compared to the K_{IA} curve, does not represent a significant reduction in the margin of safety related to brittle failure.

The methodology found in Appendix G to Section XI of the ASME B&PV Code requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}) for materials in the reactor pressure vessel beltline. A material's ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term. ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the material's copper and nickel content, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

The licensee submitted ART calculations and P-T limit curves valid for up to 32 EFPY of facility operation. For the SQN2 RPV, the licensee determined that the most limiting beltline material (i.e., the one with the highest ART value) at the 1/4T and 3/4T locations was intermediate shell forging 05, fabricated from material heat number 288757 / 981057. The ART values for the limiting beltline material at the 1/4T and 3/4T locations at 32 EFPY were 142 °F and 115 °F, respectively. The neutron fluences used in the ART calculations were 1.10 X 10^{19} n/cm^2 at the 1/4T location and 0.398 X 10^{18} n/cm^2 at the 3/4T location for 32 EFPY. The chemical composition of the limiting material was 0.13 weight percent copper and 0.76 weight percent nickel, which yields a chemistry factor from the tables in RG 1.99, Rev. 2 of 95 °F. The ΔRT_{NDT} values at the 1/4T and 3/4T locations at 32 EFPY were 97.6 °F and 70.8 °F, respectively. The initial RT_{NDT} for the SQN2 limiting beltline material was 10 °F, and was based on the availability of plant-specific data. The margin term used in calculating the ART for the limiting forging was 34 °F, consistent with the use of Position 1.1 of RG 1.99, Rev. 2 (i.e., the chemistry factor tables). Using this ART determination, the licensee then determined the material properties at these depths in the SQN2 limiting beltline material based on the application of the ASME B&PV Code static initiation fracture toughness curve (K_{IC}) referenced in ASME B&PV Code Case N-640.

Appendix G of 10 CFR Part 50 also imposes a minimum temperature requirement at the closure head flange based on the limiting reference temperature (RT_{NDT}) for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the highest reference temperature for a material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. TVA noted in their submittal that the highest RT_{NDT} value for a material in the SQN2 RPV flange region was -13 °F. Based on this value, the licensee addressed the minimum temperature requirements specified in Appendix G to 10 CFR Part 50.

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Rev. 2. Based on these calculations, the staff verified that the licensee's limiting beltline material for the SQN2 RPV is intermediate shell forging 05, fabricated from material heat number 288757 / 981057. The staff's calculated ART values for the limiting material at the 1/4T and 3/4T locations agreed with the licensee's calculated ART values.

The information submitted by the licensee regarding the material properties of the Sequoyah Unit 2 limiting beltline material and the applied loadings on the postulated flaw in the limiting beltline material was sufficient for the staff to evaluate the acceptability of the proposed SQN2 P-T limit curves.

The staff evaluated the licensee's P-T limit curves for acceptability by performing a finite set of check calculations using the methodology referenced in the ASME BP&V Code (as indicated by SRP 5.3.2) based on information submitted by the licensee. The staff verified that the licensee's proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50. Specifically, the staff concluded that the P-T limit curves submitted by the licensee were as conservative as those which would be generated by the staff's application of the methodology specified in Appendix G to Section XI of the ASME B&PV Code, as modified by ASME B&PV Code Case N-640. The staff also verified that, based on the limiting material properties of the SQN2 RPV flange, the licensee adequately addressed the minimum temperature requirements in Appendix G to 10 CFR Part 50. In a separate action, the NRC granted TVA an exemption which permits the use of ASME B&PV Code Case N-640 for the development of the SQN2 P-T limit curves.

3.2 LTOP LIMITS

The fluence values were derived from WCAP-15320. This is the report for the SQN2 surveillance capsule Y which also updates the results for capsules T, U and X. The report adheres to the guidance in RG 1.190 with respect to: cross sections, modeling, approximations and convergence. The report updated the calculations of previously analyzed surveillance capsules by repeating the analysis with updated cross sections. Comparison of the measured and corresponding calculated values indicates that the results are within the guidelines of the RG 1.190. The fluence values used in the calculation of the PT curves are the calculated values, as recommended in the regulatory guide. The NRC staff concluded that the fluence values used in the calculation of the PT curves are acceptable because the methodology adheres to the guidance in RG 1.190.

The LTOP analysis is reported in WCAP-15321, Revision 3. The analysis, which is reported in Appendix A of WCAP-15321, Revision 1, was performed in compliance with WCAP-14040-A

and includes mass and heat input transient analyses. The proposed power-operated relief valve open-setting will protect the vessel as required by Appendix G to 10 CFR Part 50 and will re-close to protect the reactor coolant pump seals from pressure undershoot. The NRC staff review indicates that the LTOP settings have been estimated using an accepted methodology and, thus, are acceptable.

As required by 10 CFR 50.61, TVA estimated the value of RT_{PTS} for the belt-line critical-element, which is the intermediate shell forging-05. The peak fluence value for 32 EFPYs is 1.8×10^{19} n/cm^2 and is reported in Reference 4. The chemistry factor is 95 and is reported in Reference 5. The 32 EFPY RT_{PTS} is 155 °F which is well within the 10 CFR 50.61 screening criterion of 270 °F and, therefore, is acceptable.

The staff concluded that the fluence values for the P-T limit curves were calculated using staff-approved methods. The staff also concluded that the LTOP analyses were performed using staff approved methods and the value of RT_{PTS} is well within the required screening criteria established in 10 CFR 50.61. Therefore, the requested LTOP limits revisions are acceptable.

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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 37583). 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 amendment.

5.0 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter from P. Salas Tennessee Valley Authority, to NRC, “Sequoyah Nuclear Plant (SQN) Unit 2 Technical Specification (TS) Change No. 03-08, Reactor Coolant System Heatup and Cooldown Curves,” June 5, 2003.

2. Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” NRC, March 2001.

3. WCAP-14040-NP-A, Revision 2, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” January 1996.

4. WCAP-15320, Revision 0, “Analysis of Capsule Y from the Tennessee Valley Authority Sequoyah Unit 2 Reactor Vessel Radiation Surveillance Program,” Westinghouse Electric Company, LLC, December 1999.

5. WCAP-15321, Revision 3, “Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation,” Westinghouse Electric Company, LLC, April 2001.

Principal Contributor: Lambros Lois, NRR
Matthew A. Mitchell, NRR

Dated: July 31, 2003

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

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

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

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

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

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

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

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

County Executive
Hamilton County Courthouse
Chattanooga, TN 37402-2801

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