October 26, 1993

Docket Nos. 50-327
and 50-328

Tennessee Valley Authority
ATTN: Dr. Mark O. Medford, Vice President
Technical Support
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M86824 AND M86825) (TS 93-07)

The Commission has issued the enclosed Amendment No. 171 to Facility Operating License No. DPR-77 and Amendment No. 161 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated June 21, 1993.

The proposed changes would revise the method of determining the most negative moderator temperature coefficient specified for end of cycle and the associated 300 ppm surveillance requirement limit specified in the Core Operating Limits Report. The TS 6.9.1.14a list of NRC-approved methodologies is also revised to include WCAP 13631.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
ORIGINAL SIGNED:
David E. LaBarge, Sr. Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 171 to License No. DPR-77
2. Amendment No. 161 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures:
See next page

*see previous concurrence

NAME:	PDII-4/LA	PDII-4/PM	OGC*	PDII-4/D	
OFFICE:	BClayton	DLaBarge	MYoung	FHebdon	
DATE:	10/25/93	10/25/93	10/21/93	10/26/93	

DOCUMENT NAME: 86824.AMM

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Tennessee Valley Authority
ATTN: Dr. Mark O. Medford

SEQUOYAH NUCLEAR PLANT

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 171
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 June 21, 1993, 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. 171, 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 within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: October 26, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 171

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
6-21	6-21

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the Updated Final Safety Analysis Report (UFSAR) analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, which is equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the UFSAR analyses to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the UFSAR safety analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions; and (2) subtracting from this value the largest differences in MTC observed between end of life (EOL), all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the UFSAR safety analyses into the limiting EOL MTC value. The 300-ppm surveillance limit MTC value represents a conservative MTC value at a core condition of 300-ppm equilibrium boron concentration, and is obtained by making these corrections for burnup and soluble boron to the limiting EOL MTC value.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly principally because of the reduction in RCS boron concentration associated with fuel burnup.

BASES

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in a OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or Safety Valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.14 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference Limits for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, K(z), and W(z) for Specification 3/4.2.2, and

6. Nuclear Enthalpy Hot Channel Factor and Power Factor Multiplier for Specification 3/4.2.3.

6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 -Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION", JUNE 1983 (W Proprietary).
   (Methodology for Specification 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(z) surveillance requirements for F_Q Methodology).)

3. WCAP-10266-P-A Rev. 2, "THE 1981 REVISION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

4. WCAP-13631-P-A, "SAFETY EVALUATION SUPPORTING A MORE NEGATIVE EOL MODERATOR TEMPERATURE COEFFICIENT TECHNICAL SPECIFICATION FOR THE SEQUOYAH NUCLEAR PLANTS," MARCH 1993 (W Proprietary).
   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
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 21, 1993, 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. 161, 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 within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: October 26, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 161

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
6-22	6-22

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the Updated Final Safety Analysis Report (UFSAR) analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, which is equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the UFSAR analyses to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the UFSAR safety analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions; and (2) subtracting from this value the largest differences in MTC observed between end of life (EOL), all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the UFSAR safety analyses into the limiting EOL MTC value. The 300-ppm surveillance limit MTC value represents a conservative MTC value at a core condition of 300-ppm equilibrium boron concentration, and is obtained by making these corrections for burnup and soluble boron to the limiting EOL MTC value.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly principally because of the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within it analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in a OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

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The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or Safety Valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.14 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference Limits for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, K(z), and W(z) for Specification 3/4.2.2, and
6. Nuclear Enthalpy Hot Channel Factor and Power Factor Multiplier for Specification 3/4.2.3.

6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 -Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Hot Channel Factor.)
2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION", JUNE 1983 (W Proprietary).
   (Methodology for Specification 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(z) surveillance requirements for F_Q Methodology).)
3. WCAP-10266-P-A Rev. 2, "THE 1981 REVISION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).
4. WCAP-13631-P-A, "SAFETY EVALUATION SUPPORTING A MORE NEGATIVE EOL MODERATOR TEMPERATURE COEFFICIENT TECHNICAL SPECIFICATION FOR THE SEQUOYAH NUCLEAR PLANTS," MARCH 1993 (W Proprietary).
   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient)

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By application dated June 21, 1993, the Tennessee Valley Authority (TVA or the licensee) proposed amendments to the Technical Specifications (TS) for Sequoyah Nuclear Plant (SQN) Units 1 and 2. The requested changes would revise the method of determining the most negative moderator temperature coefficient (MTC) for end of cycle (EOC) and the associated 300 ppm surveillance requirement (SR) limits specified in the Core Operating Limits Report (COLR). The purpose of the 300 ppm SR is to ensure that the most negative MTC at EOC remains within the bounds of the SQN Units 1 and 2 safety analyses, in particular for those transients and accidents that assume a constant value of the moderator density coefficient (MDC) of 0.43 delta-K per gm/cc.

The SQN SR involves an MTC measurement at any thermal power within 7 effective full power days (EFPD) after reaching an equilibrium primary coolant boron concentration of 300 ppm. After appropriate corrections are made, the measured value is compared to the 300 ppm SR limit value specified in the COLR at the all-rods-out (ARO) rated thermal power (RTP) condition. In the event that the measured MTC is more negative than the 300 ppm SR limit, the MTC must be remeasured and compared with the EOC MTC limiting condition of operation (LCO) value at least once per 14 EFPD during the remainder of the operating cycle. The 300 ppm SR and EOC LCO values for the most negative MTC are conservative (less negative) when compared to the value of the MTC used in the safety analyses.

SQN also proposed to revise the current method for determining the 300-ppm surveillance and the EOL MTC limits specified in the COLR. The revised method for determining the COLR MTC limits will result in a change to the TS Bases Section B 3/4.1.1.3. This revised method and the COLR MTC limit changes do not affect the maximum (MDC) value of 0.43 delta k/gm/cc, which corresponds to an MTC value of -52.68 pcm/^{o}F. These changes apply to the current and future reload cycles for SQN Units 1 and 2, and are supported by a NRC approved Westinghouse Methodology evaluation (WCAP 13631). This evaluation applies only to SQN and is similar to that approved for use at other nuclear power plants.

A proposed change to TS 6.9.1.14.a would add a reference to WCAP 13631 as one of the documents that contain the analytical methods used to determine the core operating limits. The staff's review of these proposed changes follows.

2. EVALUATION

2.1 Methodology

2.1.1 Amendment to End of Cycle Moderator Temperature Coefficient

The current method used to determine the most negative MTC is described in Bases Section B 3/4.1.1.3 of the TS for SQN Units 1 and 2. This method is based on incrementally correcting the conservative MDC used in the safety analyses to obtain the most negative MTC value or, equivalently, the most positive MDC at the nominal hot full power (HFP) core conditions. The corrections involve subtracting the incremental change in the MDC, which is associated with the core condition of all control rods inserted (ARI), to an ARO core condition. The MTC is then equal to the product of multiplying the MDC by the rate of change of the moderator density with the temperature at rated thermal power (RTP) conditions.

The TS Bases provide a method of determining the most negative MTC LCO value that results in an ARO MTC value that is significantly less negative than the MTC used in the safety analysis and that may be less negative than the best estimate EOC ARO MTC for extended burnup reload cores. This could result in TS 3.1.1.3 specifying that the plant be placed in a hot shutdown condition even though it would retain a substantial margin to the safety analysis MDC. The problem with the current method is caused by adjusting the MDC from an HFP ARI condition to an HFP ARO condition in defining the most negative MTC. The TS on control rod positions does not allow the HFP ARI condition for allowable power operation in which the shutdown banks are completely withdrawn from the core and the control banks meet the rod insertion limits (RILs).

The licensee provided an alternative method for adjusting the safety analysis MDC to obtain a most negative MTC. The "most negative feasible" (MNF) method seeks to determine the conditions for which a core will exhibit the most negative value that is consistent with operation allowed by the TS. The licensee uses the MNF method to determine EOC MTC sensitivities to those design and operational parameters that directly affect the MTC in such a way that the sensitivity to one parameter depends on the assumed values for the other parameters.

The licensee stated that this MNF MTC approach has a number of advantages over the previous method for determining the most negative MTC LCO value. The MNF MTC would be sufficiently negative that repeated MTC measurements from a concentration of 300 ppm of boron in the core to EOC would not be required. The MNF MTC method does not change the moderator feedback assumption or the value of the MDC in the safety analysis. The MNF MTC method is a conservative and reasonable basis to assume for an MTC value of a reload core and is consistent with plant operation defined by other TS. Finally, the MNF MTC method retains the SR on MTC at the 300 ppm core condition to verify that the core is operating within the bounds of the safety analysis.

The licensee stated that the SR MTC value would be obtained in the same manner as currently described in the Westinghouse Standard Technical Specification (STS) Bases. The SR MTC value is obtained from the EOC ARO MTC value by making corrections for burnup and boron at a core condition of 300 ppm of boron.

The staff review of the assumptions and basis for the MNF MTC method described in this safety evaluation concluded that they are acceptable because they will result in conservative (more negative) MTC SR and EOC values during allowed operation of SQN Units 1 and 2 at nominal conditions and because the MTC measurement at 300 ppm of boron core condition will ensure, using the SR value of MTC, that the safety analysis MDC will not be exceeded.

After evaluating the proposed changes to the TS, the staff has determined that the proposed change to the TS Bases method of determining the EOC MTC and 300 ppm SR limit values specified in the COLR is acceptable for the following reasons:

(1) The most negative feasible MTC method considered the important factors affecting the MTC and the limits on the these factors.

(2) Westinghouse used approved methods and computer codes in the analysis.

(3) Measuring the MTC at or near 300 ppm of boron will ensure that the MTC at EOC HFP ARO conditions will be less negative than the safety analysis.

(4) The licensee will analyze future reloads for SQN Units 1 and 2 to confirm the most negative MTC TS at EOC and SR on MTC at a core condition of 300 ppm of boron.

(5) The licensee will analyze future reloads for SQN Units 1 and 2 to confirm that the safety analysis value of the MDC applies.

(6) Upon receipt of this amendment, the licensee will revise the current COLR as appropriate and submit the revised COLR to the staff as required by TS 6.9.1.14.c.

The licensee also proposes to change to TS 6.9.1.14.a by adding a reference to the Westinghouse document (WCAP 13631) that describes the analytical methods used to implement the revised methodology. This change is in accordance with Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," and will assure that future reloads will meet all applicable limits and acceptance criteria of the safety analysis, that the cycle-specific parameter limits will continue to be determined in accordance with NRC-approved methodology, and that operation will be limited/consistent with 10 CFR 50.36(c). Accordingly, the staff finds the changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The amendment also changes recordkeeping or reporting 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 (58 FR 41515). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10). 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.

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

Principal Contributor: A. Attard

Dated: October 26, 1993

AMENDMENT NO. 171 FOR SEQUOYAH UNIT NO. 1 - DOCKET NO. 50-327 and
AMENDMENT NO. 161 FOR SEQUOYAH UNIT NO. 2 - DOCKET NO. 50-328
DATED: October 26, 1993

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