November 19, 1998

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: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M96594 AND M96595) (TS 96-03)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 238 to Facility Operating License No. DPR-77 and Amendment No. 228 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively. These amendments are in response to your application dated August 21, 1996. The amendments revise the SQN Technical Specification 3.7.1.3 to extend the limiting condition for operation of the condensate storage tanks to Mode 4 when steam generators are relied upon for heat removal. The U.S. Nuclear Regulatory Commission staff has found your proposed changes to be acceptable.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. Please direct any questions you or your staff should have to me at 301-415-2010.

Sincerely,

Original signed by:

Ronald W. Hernan, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 238 to License No. DPR-77
2. Amendment No. 228 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

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

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The Commission has issued the enclosed Amendment No. 238 to Facility Operating License No. DPR-77 and Amendment No. 228 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively. These amendments are in response to your application dated August 21, 1996. The amendments revise the SQN Technical Specification 3.7.1.3 to extend the limiting condition for operation of the condensate storage tanks to Mode 4 when steam generators are relied upon for heat removal. The U.S. Nuclear Regulatory Commission staff has found your proposed changes to be acceptable.

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Sincerely,

Ronald W. Hernan

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Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 238 to License No. DPR-77
2. Amendment No. 228 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

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. 238
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 August 21, 1996, 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. 238 , 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 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 19, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 238

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
3/4 7-7	3/4 7-7
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 A condensate storage tank system (CST) shall be OPERABLE with a contained water volume of at least 190,000 gallons of water. FP

APPLICABILITY: MODES 1, 2 and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTION:

With the condensate storage tank system inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours without reliance on steam generator for heat removal, or

b. Verify by administrative means OPERABILITY of the Essential Raw Cooling Water System as a backup supply to the auxiliary feedwater pumps* and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours without reliance on steam generator for heat removal.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank system shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

* OPERABILITY shall be verified once per 12 hours following initial verification.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable in the updated final safety analysis report. R194

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

Action 15 of Table 3.3-1, Reactor Trip System Instrumentation, allows the breaker to be bypassed for up to 4 hours for the purpose of performing maintenance. The 4 hours is based on a Westinghouse analysis performed in WCAP-10271, Supplement 1, which determines bypass breaker availability. R58

The placing of a channel in the trip condition provides the safety function of the channel. If the channel is tripped for testing and no other condition would have indicated inoperability, the channel should not be declared inoperable. BR-9

The Auxiliary Feedwater (AFW) Suction Pressure-Low function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the steam generators as the heat sink for the reactor. This function does not have to be OPERABLE in MODES 5 and 6 because heat being generated in the reactor is removed via the Residual Heat Removal (RHR) System and does not require the steam generators as a heat sink. In MODE 4, AFW automatic suction transfer does not need to be OPERABLE because RHR will already be in operation, or sufficient time is available to place RHR in operation to remove decay heat.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(X,Y,Z)$ or $F_{\Delta H}(X,Y)$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable. R227

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974. R85

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility and the potential capability for subsequent cold shutdown from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50. BR

3/4.3.3.6 CHLORINE DETECTION SYSTEMS

This specification deleted. R66

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980. R153

The postaccident monitoring instrumentation limiting condition for operation provides the requirement of Type A and Category 1 monitors that provide information required by the control room operators to: R163

- Permit the operator to take preplanned manual actions to accomplish safe plant shutdown.

- Determine whether systems important to safety are performing their intended functions.

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. 228
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 August 21, 1996, 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. 228 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 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments: 1.Changes to the Technical Specifications

Date of Issuance: November 19, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 228

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
3/4 7-7	3/4 7-7
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank system (CST) shall be OPERABLE with a contained water volume of at least 190,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTION:

With the condensate storage tank system inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours without reliance on steam generator for heat removal, or

b. Verify by administrative means OPERABILITY of the essential raw cooling water system as a backup supply to the auxiliary feedwater pumps* and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours without reliance on steam generator for heat removal.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank system shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the system is the supply source for the auxiliary feedwater pumps.

* OPERABILITY shall be verified once per 12 hours following initial verification.

REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the protective and the engineered safety feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable in the updated final safety analysis report. R182

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

Action 15 of Table 3.3-1, Reactor Trip System Instrumentation, allows the breaker to be bypassed for up to 4 hours for the purpose of performing maintenance. The 4 hours is based on a Westinghouse analysis performed in WCAP-10271, Supplement 1, which determines bypass breaker availability. R46

The placing of a channel in the trip condition provides the safety function of the channel. If the channel is tripped for testing and no other condition would have indicated inoperability, the channel should not be declared inoperable. BR-10

The Auxiliary Feedwater (AFW) Suction Pressure-Low function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the steam generators as the heat sink for the reactor. This function does not have to be OPERABLE in MODES 5 and 6 because heat being generated in the reactor is removed via the Residual Heat Removal (RHR) System and does not require the steam generators as a heat sink. In MODE 4, AFW automatic suction transfer does not need to be OPERABLE because RHR will already be in operation, or sufficient time is available to place RHR in operation to remove decay heat.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(X,Y,Z)$ or $F_{\Delta H}(X,Y)$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable. R214

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974. R72

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility and the potential capability for subsequent cold shutdown from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50. BR

3/4.3.3.6 CHLORINE DETECTION SYSTEMS R54

This specification deleted.

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980. R135

The postaccident monitoring instrumentation limiting condition for operation provides the requirement of Type A and Category 1 monitors that provide information required by the control room operators to: R149

- Permit the operator to take preplanned manual action to accomplish safe plant shutdown

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 238 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 228 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

The Tennessee Valley Authority (TVA, the licensee) requested amendments to Operating Licenses DPR-77 and DPR-79 for Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively, in a letter to the U.S. Nuclear Regulatory Commission (NRC) dated August 21, 1996. The amendments would revise the SQN Units 1 and 2 Technical Specifications (TS) to extend the applicability of the limiting condition of operation (LCO) for the condensate storage tank (CST) to Mode 4 when the steam generators are relied upon for heat removal. The licensee also proposed changes to associated action statements, surveillance requirements (SR), and TS bases.

2.0 BACKGROUND

In 1995, by license Amendment Nos. 206 and 196 for Units 1 and 2, respectively, the licensee revised TS 3.7.1.2, Auxiliary Feedwater System, to be consistent with the Westinghouse Standard Technical Specifications contained in NUREG-1431. This change expanded the mode applicability for the auxiliary feedwater (AFW) system's LCO from Modes 1, 2, and 3, to include "MODE 4 when steam generator is relied upon for heat removal." At that time, the licensee did not recognize the need for a change to the mode of applicability for the CST which provides the water source for the AFW pumps. Presently, TS 3.7.1.3 requires the CST to be operable in Modes 1, 2, and 3. This is not consistent with the TS required AFW operability in Mode 4. As an interim corrective action, the licensee is currently operating with administrative controls to ensure that CST is operable in all applicable modes when the AFW system is required to remain operable. The licensee has now proposed appropriate TS changes to correct the disparity between the CST and AFW TS.

TVA proposes to modify TS 3.7.1.3, "Condensate Storage Tank" to:

(a) extend the mode of applicability to include "MODE 4 when steam generator is relied upon for heat removal." Current modes of applicability are Modes 1, 2, and 3.

(b) revise Action (a) to increase the allowable action time for achieving hot shutdown to 12 hours from the current 6 hours, and to add the phrase "without reliance on steam generator for heat removal "to be consistent with the language in NUREG-1431.

(c) revise Action (b) to modify the phrase "Demonstrate the operability Essential Raw Cooling Water System as a backup supply to the auxiliary feedwater pumps ..." to "Verify by administrative means operability of the Essential Raw Cooling Water System (ERCW) as a backup supply to the auxiliary feedwater pumps ..." to be consistent with the language in NUREG-1431.

(d) add the phrase "without reliance on steam generator for heat removal" to Action (b) to be consistent with the language in NUREG-1431.

(e) delete the current SR 4.7.1.3.2 for verifying ERCW operability every 12 hours and make the requirement a footnote to Action (b) instead.

(f) revise TS Bases 3/4.3.2 to provide guidance concerning the AFW suction pressure-low function as to the modes when this function is required to be operable.

(g) correct a typographical error in Action (b) by replacing "pmps" with "pumps."

3.0 EVALUATION

Following a reactor trip, decay heat is dissipated by evaporating water in the steam generators and venting the steam either to the condensers or to the atmosphere. In such situations, steam generator water inventory must be maintained at a level sufficient to ensure adequate heat transfer and decay heat removal. The AFW system pumps deliver this emergency water supply to the steam generators. The AFW system provides emergency water to the steam generators until either normal feed water flow is established or the residual heat removal (RHR) system can assume the decay heat removal function. The primary sources of water for the AFW system pumps are the condensate storage tanks. On low suction pressure, the AFW pumps are designed to automatically swap to the ERCW system.

SQN has two CSTs (one for each unit) that provide cooling water to the suction of the AFW pumps. TVA's proposed change to extend the mode of applicability for the CST to "MODE 4 when steam generator is relied upon for heat removal" would ensure that a source of water is available to the required AFW train until the RHR system can assume the decay heat removal function. This change is consistent with the mode requirements for AFW operability. Therefore, the staff finds the proposed change to the CST operability acceptable.

Adding the phrase "without reliance on steam generator for heat removal" to current Actions (a) and (b) would be done for the pupose of ensuring unit cooldown to RHR entry conditions without reliance on steam generators. The staff notes that this phase is consistent with the language in Section 3.7.6 of NUREG-1431 and is discussed in Bases Section B 3.7.6. The staff finds the proposed change to be consistent with the proposed mode of applicability for the CST and is, therefore, acceptable. The staff also finds the action time for achieving hot standby from 6 hours to 12 hours is reasonable for transition from the steam generator mode of decay heat removal to RHR entry conditions and would achieve the required plant condition in an orderly manner without challenging other plant systems.

SR 4.7.1.3.2 for demonstrating operability of the ERCW system is redundant to TS 3.7.4 which requires two ERCW loops to be operable in Modes 1, 2, 3, and 4. SR 4.7.1.3.2 for demonstrating operability of the ERCW system is, therefore, not necessary and the staff finds deleting SR 4.7.1.3.2 to be acceptable.

The staff also finds the proposed TS changes discussed in Items (c), (e), and (g) above to be editorial and, therefore, acceptable.

4.0 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.

5.0 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 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 52967, dated October 9, 1996). 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 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.

Principal Contributor: L. Raghavan, NRR

Date: November 19, 1998