May 11, 1989

Docket Nos. 50-327, 50-328

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Mr. Oliver D. Kingsley, Jr.
Senior Vice President, Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: AUXILIARY FEEDWATER PUMP DIFFERENTIAL PRESSURE OPERABILITY (TAC 72209/R00353) (TS 88-24/88-02) - SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

The Commission has issued the enclosed Amendment No. 115 to Facility Operating License No. DPR-77 and Amendment No. 105 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 February 23, 1989.

The amendments modify the Sequoyah Nuclear Plant, Units 1 and 2, Technical Specifications (TS). The changes revise the surveillance requirement 4.7.1.2.a to add specific differential pressure test values for each auxiliary feedwater (AFW) pump. The associated bases section is revised to clarify the AFW technical specification requirements.

The changes for the Unit 2 TS superseded the values submitted in your letter dated May 26, 1988 for TS change number 88-02. The new higher proposed differential pressure values for Unit 2 are to provide additional margin to offset uncertainties in the flow and pressure test data for the three AFW pumps. A revised bases section for the Unit 2 change was also submitted.

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

Sincerely,
Original signed by
Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 115 to License No. DPR-77
2. Amendment No. 105 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

# TENNESSEE VALLEY AUTHORITY

## DOCKET NO. 50-327

## SEQUOYAH NUCLEAR PLANT, UNIT 1

## AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 115
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 February 23, 1989, 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. 115, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 11, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 115

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. Overleaf pages* are provided to maintain document completeness.

| REMOVE | INSERT |
| --- | --- |
| 3/4 7-5 | 3/4 7-5 |
| 3/4 7-6 | 3/4 7-6* |
| B 3/4 7-1 | B 3/4 7-1* |
| B 3/4 7-2 | B 3/4 7-2 |

## AUXILIARY FEEDWATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate shutdown boards, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

### SURVEILLANCE REQUIREMENTS

4.7.1.2 In addition to the requirements of Specification 4.0.5 each auxiliary feedwater pump shall be demonstrated OPERABLE by :

a. Verifying that:

1. each motor-driven pump develops a differential pressure of greater than or equal to the values indicated below on recirculation flow.

   1A-A: 1450 psid.

   1B-B: 1500 psid.

2. the steam turbine-driven pump develops a differential pressure of greater than or equal to 1201 psid on recirculation flow when the secondary steam supply pressure is greater than 842 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

3. at least once per 31 days, each automatic control valve in the flow path is OPERABLE whenever the auxiliary feedwater system is placed in automatic control or when above 10% of RATED THERMAL POWER.

b. At least once per 18 months during shutdown* by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal and a low auxiliary feedwater pump suction pressure test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an auxiliary feedwater actuation test signal.

c. At least once per 7 days by verifying that each non-automatic valve in the auxiliary feedwater system flowpath is in its correct position.

*The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven Auxiliary Feedwater Pump.

## 3/4.7.1 TURBINE CYCLE

### 3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of the system design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is $1.9 \times 10^7$ lbs/hr at 1170 psig which is 127 percent of the total secondary steam flow of $1.493 \times 10^7$ lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (76)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

U = maximum number of inoperable safety valves per operating steam line.

109 = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation.

76 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 3 loop operation.

X = Total relieving capacity of all safety valves per steam line in lbs/hour, $4.75 \times 10^6$ lbs/hour at 1170 psig.

Y = Maximum relieving capacity of any one safety valve in lbs/hour, 950,000 lbs/hour at 1170 psig.

## 3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

The steam driven auxiliary feedwater pump is capable of delivering 880 gpm (total feedwater flow) and each of the electric driven auxiliary feedwater pumps are capable of delivering 440 gpm (total feedwater flow) to the entrance of the steam generators at steam generator pressures of 1100 psia. At 1100 psia the open steam generator safety valve(s) are capable of relieving at least 11% of nominal steam flow. A total feedwater flow of 440 gpm at pressures of 1100 psia is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F where the Residual Heat Removal System may be placed into operation. The surveillance test values ensure that each pump will provide at least 440 gpm plus pump recirculation flow against a steam generator pressure of 1100 psia.

## 3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 2 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usuable because of tank discharge line location or other physical characteristics.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
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 February 23, 1989, 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. 105, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 11, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 105

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. Overleaf pages* are provided to maintain document completeness.

| REMOVE | INSERT |
|---|---|
| 3/4 7-5 | 3/4 7-5 |
| 3/4 7-6 | 3/4 7-6* |
| B 3/4 7-1 | B 3/4 7-1* |
| B 3/4 7-2 | B 3/4 7-2 |

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

## LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate shutdown boards, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: Modes 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

## SURVEILLANCE REQUIREMENTS

4.7.1.2 In addition to the requirements of Specification 4.0.5 each auxiliary feedwater pump shall be demonstrated OPERABLE by:

a. Verifying that:

   1. each motor-driven pump develops a differential pressure of greater than or equal to the values indicated below on recirculation flow.

      2A-A: 1524 psid

      2B-B: 1464 psid

   2. the steam turbine-driven pump develops a differential pressure of greater than or equal to 1180 psid on recirculation flow when the secondary steam supply pressure is greater than 842 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

3. at least once per 31 days, each automatic control valve in the flow path is OPERABLE whenever the auxiliary feedwater system is placed in automatic control or when above 10% of RATED THERMAL POWER.

b. At least once per 18 months during shutdown* by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal and a low auxiliary feedwater pump suction pressure test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an auxiliary feedwater actuation test signal.

c. At least once per 7 days by verifying that each non-automatic valve in the auxiliary feedwater system flowpath is in its correct position.

*The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven Auxiliary Feedwater Pump.

## 3/4.7.1 TURBINE CYCLE

### 3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of the system design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is $1.9 \times 10^7$ lbs/hr at 1170 psig which is 127 percent of the total secondary steam flow of $1.493 \times 10^7$ lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times 76$$

Where:

SP = Reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = Maximum number of inoperable safety valves per steam line

U = Maximum number of inoperable safety valves per operating steam line

SAFETY VALUES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

76 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 3 loop operation.

X = Total relieving capacity of all safety valves per steam line in lbs/hour, $4.75 \times 10^6$ lbs/hr at 1170 psig

Y = Maximum relieving capacity of any one safety valve in lbs/hour, $9.5 \times 10^5$ lbs/hr at 1170 psig.

## 3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

The steam driven auxiliary feedwater pump is capable of delivering 880 gpm (total feedwater flow) and each of the electric driven auxiliary feedwater pumps are capable of delivering 440 gpm (total feedwater flow) to the entrance of the steam generators at steam generator pressures of 1100 psia. At 1100 psia the open steam generator safety valve(s) are capable of relieving at least 11% of nominal steam flow. A total feedwater flow of 440 gpm at pressures of 1100 psia is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F where the Residual Heat Removal System may be placed into operation. The surveillance test values ensure that each pump will provide at least 440 gpm plus pump recirculation flow against a steam generator pressure of 1100 psia.

## 3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 2 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

## 3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# SUPPORTING AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. DPR-77

# AND AMENDMENT NO. 105 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

In letters dated May 26, 1988 and February 23, 1989, the Tennessee Valley Authority (TVA) requested changes in the Technical Specifications (TS) for Sequoyah Nuclear Plant (SQN), Units 1 and 2, regarding the acceptance criteria for auxiliary feedwater pump (AFWP) differential pressure surveillance testing. Specifically, the proposed changes revised Surveillance Requirement (SR) 4.7.1.2.a for each unit's TS to add pump-specific differential pressure test values for each AFWP. These changes were necessitated by modifications to the AFWPs and/or their associated piping flowpaths. In addition, the bases sections associated with these SRs were revised by the proposed TS changes. The proposed TS change is number 88-02 for Unit 2 and 88-24 for Unit 1.

The staff developed concerns, with the May 26, 1988 submission of TS change number 88-02 for Unit 2, relating to the method used to incorporate test instrument error in the generation of the AFW pump curves, system resistance curves, and SR 4.7.1.2.a acceptance criteria. In response to the staff concerns, TVA reevaluated their calculations and provided revised proposed TS change pages for SR 4.7.1.2.a and its associated bases. These revised pages, along with the corresponding calculation revision, were forwarded as additional information as part of the February 23, 1989 request for TS change number 88-24 for SQN Unit 1.

The TVA letter dated February 23, 1989 applied for changes on the AFWP for the Unit 1 TS. It also applied for changes on the AFWP for the Unit 2 TS. The latter changes superseded the proposed changes in the TVA letter dated May 26, 1988. In the Federal Register Notice of consideration of issuance of an amendment (54 FR 11844) published on March 22, 1989 for TVA application dated February 23, 1989, it was acknowledged that the May 26, 1988 application was being superseded and the staff's initial determination of no significant hazards consideration was for the entire application dated February 23, 1989.

## 2.0 EVALUATION

TVA replaced the pressure control valves in the AFWP discharge lines with cavitating venturis to improve system performance. These modifications changed the system resistance of the AFW flowpath, which necessitated redefinition of the minimum acceptable pump differential pressures for the AFWP operability determinations required by SR 4.7.1.2.a. To develop the new acceptance criteria, startup testing (in Mode 3) of the AFW system was required. Because the units were in cold shutdown (Mode 5), TVA was unable to conduct the testing necessary to determine the SR acceptance values. TVA submitted a startup testing plan for NRC approval by letters dated December 23, 1986, and June 25 and September 24, 1987. This plan described the general methodology to be followed in the startup test program, and requested NRC approval to enter Mode 3 in order to conduct the necessary AFW testing at the proper plant conditions. TVA's plan was approved on November 2, 1987. Besides allowing entry into Mode 3, this approval included calculated allowable pump degradation values to be used in conjunction with SR 4.7.1.2.a until a TS change, developed from the results of the startup testing, could be approved by the staff.

The Mode 3 startup testing was performed at normal AFW system operating pressures and flows in order to determine the actual post modification system resistance. Once the AFW piping headloss was determined, TVA determined the minimum required AFWP discharge head to establish the required flowrate. The maximum allowable pump discharge pressure degradation was determined by subtracting the minimum required head from the actual pump head. The allowable pressure degradation was then subtracted from the nominal pump discharge pressure with the pump discharges aligned to the recirculation flowpath, which is the configuration used in the conduct of SR 4.7.1.2.a, resulting in the lowest allowable differential pressure to demonstrate AFWP operability.

During post-modification testing in 1985, TVA discovered that AFWP 2A-A was not performing in accordance with the manufacturer's pump performance curve. TVA replaced the impeller in AFWP 2A-A and generated a new pump performance curve based on test data. This new curve was incorporated into the calculations performed to establish the new AFWP 2A-A differential pressure criteria.

TVA performed Nuclear Engineering (NE) calculation 2219280000 based on the startup testing results for SQN Unit 2. This calculation was submitted in TVA's letter dated May 26, 1988 as supporting documentation for the Unit 2 TS change request. NRC staff developed concerns regarding the method used to account for test instrumentation inaccuracies in the calculation, and with the clarity of the revised TS Bases Section 3/4.7.1.2 associated with SR 4.7.1.2.a.

In response to the staff concerns, TVA reevaluated the method used to address testing instrumentation error for the Unit 2 submission, and incorporated their solutions to the NRC concerns in developing the equivalent submission for SQN Unit 1. The revision to NE calculation 2219280000, which included SR acceptance criteria determinations for both Units 1 and 2, was included as an enclosure to

TVA's February 23, 1989, letter requesting approval of TS change 88-24 for Unit 1 and 88-02 for Unit 2. All NRC staff concerns regarding TS change 88-02 in the application dated May 26, 1988 were satisfactorily resolved by the February 23, 1989 letter, and no new concerns were developed by the staff regarding either TS change request.

In reviewing the methodology, calculations and procedures associated with this TS change, the staff assessed TVA's engineering analysis for assurance that the AFWPs are adequate, with margin, to meet the steam generator flow and pressure requirements under worst-case conditions. The required conditions stated in the Final Safety Analysis Report (FSAR), Section 10.4.7.2, are flow of 440 gpm to the steam generators against a pressure of 1100 psia.

The new minimum AFWP differential pressures for SR 4.7.1.2.a are:

| | | |
|---|---|---|
| Motor Driven AFWPs: | 1A-A* | 1450 psid |
| | 1B-B | 1500 psid |
| | 2A-A | 1524 psid |
| | 2B-B | 1464 psid |
| Turbine Driven AFWPs: | 1A-S* | 1201 psid |
| | 2A-S | 1180 psid |

The individual AFWP differential pressures tabulated above reflect the pump curves and flow paths associated with each pump. These proposed acceptance criteria will demonstrate that the pumps will deliver at least 440 gpm, plus recirculation flow, at steam generator pressures of 1100 psia, therefore, ensuring that plant operation will remain bounded by the assumptions for AFW flow in the FSAR analyses.

The revisions made to Section 3/4.7.1.2 of the bases clarify the technical specification requirements and design bases for the AFW system.

Because the revised SR 4.7.1.2.a ensures conformance with the FSAR accident analysis assumptions, the probability or consequences of an accident previously evaluated remain unchanged. No changes, other than those to the testing values of the AFWP, are made to the AFW system by this proposed change. Therefore the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed. Finally, because AFW operation remains bounded by the FSAR analysis, there is no reduction in the overall plant margin of safety.

## 3.0 CONCLUSION

Based on the above, the staff has concluded that TVA's calculations and engineering analysis support the proposed changes to SR 4.7.1.2.a, and Bases Section 3/4.7.1.2, of the Sequoyah TS. The staff concludes that the TS changes numbers 88-02 and 88-04 in TVA's application dated February 23, 1989 are acceptable.

* 1 = Unit 1, 2 = Unit 2

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to 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 to the surveillance requirements. The 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. 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 nor environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 11844) on March 22, 1989 and consulted with the State of Tennessee. No public comments were received and the State of Tennessee did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: P. Castleman

Dated: May 11, 1989