Mr. Robert G. Byram
Senior Vice President-Generation
and Chief Nuclear Officer
PP&L, Inc.
2 North Ninth Street
Allentown, PA 18101

February 17, 1999

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 (TAC NO. MA2434)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 154 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 4, 1998, as supplemented by letters dated December 16, 1998, January 12, 1999, and January 28, 1999.

This amendment would modify the references in TS Section 5.6.5 of a critical power correlation applicable to Siemens Power Corporation Atrium-10 fuel and would include a revised minimum critical power ratio safety limit in TS Section 2.1.1.2.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,

Original signed by:

Victor Nerses, Senior Project Manager
Project Directorate I-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-388

Enclosures: 1. Amendment No. 154 to License No. NPF-22
2. Safety Evaluation

cc w/encls: See next page

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NAME	VNerses/rst		MO'Brien		SBajwa		SE dated		MYoung	
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NAME	VNerses/rst		MO'Brien		SBajwa		SE dated		MYoung	
DATE	02/17/99		02/11/99		02/17/99		01/20/99		02/16/99	

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

February 17, 1999

Mr. Robert G. Byram
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Victor Nerses

Victor Nerses, Senior Project Manager
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-388

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License No. NPF-22
2. Safety Evaluation

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

PP&L, INC.

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
License No. NPF-22

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by PP&L, Inc., dated August 4, 1998, as supplemented by letters dated December 16, 1998, January 12, 1999, and January 28, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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.

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P PDR

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 the Facility Operating License No. NPF-22 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 154 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and is to be implemented within 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

SBajwa

S. Singh Bajwa, Director
Project Directorate I-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 17, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 154

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following pages of the Appendix A Technical Specifications with enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
2.0 - 1	2.0 - 1
2.0 - 2	2.0 - 2
2.0 - 3	2.0 - 3
5.0 - 21	5.0 - 21
5.0 - 22	5.0 - 22
5.0 - 23	5.0 - 23
5.0 - 24	5.0 - 24
5.0 - 25	5.0 - 25
B 2.0 - 2	B 2.0 - 2
B 2.0 - 3	B 2.0 - 3
B 2.0 - 4	B 2.0 - 4
B 2.0 - 5	B 2.0 - 5
B 3.2 - 5	B 3.2 - 5
B 3.2 - 6	B 3.2 - 6
B 3.2 - 9	B 3.2 - 9

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10 million lbm/hr:

MCPR shall be ≥ 1.11 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(Figure 2.1.1.2-1)

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(Figure 2.1.1.2-2)

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5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4 The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and

5. The Shutdown Margin for Specification 3.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

2. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc. June 1986.

(continued)

5.6 Reporting Requirements

3. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, "Exxon Nuclear Company, Inc., September 1986.

4. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1, 2, and 3, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc., March 1983.

5. ANF-524(P)(A), Revision 2 and Supplement 1, Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors", November 1990.

6. ANF-1125(P)(A) and ANF-1125(P)(A), Supplement 1, "ANFB Critical Power Correlation", April 1990.

7. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

8. NE-092-001A, Revision 1, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992.

9. NRC SER on PP&L Power Uprate LTR (November 30, 1993).

10. PL-NF-90-001, Supplement 1-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Loss of Feedwater Heating Changes and Use of RETRAN MOD 5.1," August 1995.

11. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES", January, 1995.

12. NEDE-24011-P-A-10, "General Electric Standard Application For Reactor Fuel, February, 1991.

13. PL-NF-90-001, Supplement 2-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation", July 1996.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

14. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

15. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

16. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.

17. XN-NF-80-19(P)(A), Volumes 3 Revision 2 "Exxon Nuclear Methodology for Boiling Water Reactors Thermex: Thermal Limits Methodology Summary Description," January 1987.

18. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

19. EMF-1997 (P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

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5.6 Reporting Requirements

5.6.5 COLR (continued)

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(continued)

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

The use of the ANFB (Reference 2) and ANFB-10 (Reference 4) correlations are valid for critical power calculations at pressures > 600 psia for ANFB and > 571 psia for ANFB-10 and bundle mass fluxes > 0.1×10^6 lb/hr-ft^2 for ANFB and > 0.115×10^6 lb/hr-ft^2 for ANFB-10. For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the SPC 9x9 fuel design, the minimum bundle flow is approximately 30×10^3 lb/hr. For the SPC Atrium 10 design,

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

the minimum bundle flow is > 28 x 10^3 lb/hr. For both the SPC 9x9-2 and Atrium-10 fuel designs, the coolant minimum bundle flow and maximum area are such that the mass flux is always > .25 x 10^6 lb/hr-ft^2. Full scale critical power test data taken from various SPC and GE fuel designs at pressures from 14.7 psia to 1400 psia indicate the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of approximately 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty in the ANFB critical power correlation. Reference 2 describes the methodology used in determining the MCPR SL.

The ANFB and ANFB-10 critical power correlations are based on a significant body of practical test data. As long as the core pressure and flow are within the range of validity of the correlation (refer to Section B 2.1.1.1), the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the ANFB and ANFB-10 correlations provide a reasonable degree of assurance that during sustained operation at the MCPR SL there would be no transition boiling in the core. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

SPC 9x9-2 fuel is monitored using the ANFB critical power correlation, and the SPC ATRIUM-10 fuel is monitored using the ANFB-10 Critical Power Correlation. The effects of channel bow on MCPR are explicitly included in the calculation of the MCPR SL. Explicit treatment of channel bow in the MCPR SL addresses the concerns of the NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow." The Unit 2 core contains four GE lead use assemblies (LUAs). The LUAs are loaded in nonlimiting core regions per Specification 4.2.1. The MCPR SL generated using Reference 2 is acceptable for the GE LUAs.

Monitoring required for compliance with the MCPR SL is specified in LCO 3.2.2, Minimum Critical Power Ratio.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

(continued)

BASES (continued)

SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. ANFB 524 (P)(A), Revision 2, "Critical Power Methodology for Boiling Water Reactors," Supplement 1 Revision 2 and Supplement 2, November 1990.
3. 10 CFR 100.
4. EMF-1997, Revision 0 (October 1997) and Supplement 1, Revision 0 (January 1998), "ANFB-10 Critical Power Correlation," and associated NRC SER dated 7/17/98.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2 through 9. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

state to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency. These analyses may also consider other combinations of plant conditions (i.e., control rod scram speed, bypass valve performance, EOC-RPT, cycle exposure, etc.). Flow dependent MCPR limits are determined by analysis of slow flow runout transients using the methodology of Reference 2.

The Unit 2 core contains four GE lead use assemblies (LUAs). The LUAs are loaded in nonlimiting core regions per specification 4.2.1. MCRR operating limits for the GE LUAs have been developed using the methodology from References 2 and 11.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 10).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the flow dependent MCPR and power dependent MCPR limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the

(continued)

BASES

REFERENCES (continued)

3. PL-NF-87-001-A, "Qualification of Steady State core Physics Methods for BWR Design and Analysis," April 28, 1988.

4. PL-NF-89-005-A, "Qualification of Transient Analysis Methods for BWR Design and Analysis," July 1992, including Supplements 1 and 2.

5. XN-NF-80-19 (P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

6. NE-092-001, Revision 1, "Susquehanna Steam Electric Station Units 1 & 2: Licensing Topical Report for Power Uprate with Increased Core Flow," December 1992, and NRC Approval Letter: Letter from T. E. Murley (NRC) to R. G. Byram (PP&L), "Licensing Topical Report for Power Uprate With Increased Core Flow, Revision 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC Nos. M83426 and M83427)," November 30, 1993.

7. EMF-1997, Revision 0 (October 1997) and Supplement 1, Revision 0 (January 1998), "ANFB-10 Critical Power Correlation," and associated NRC SER dated 7/17/98.

8. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

9. XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987.

10. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

11. NEDE-24011-P-A-10, "General Electric Standard Application for Reactor Fuel," February 1991.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 154 TO LICENSE NO. NPF-22

PENNSYLVANIA POWER AND ELECTRIC COMPANY

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-388

1.0 INTRODUCTION

By letter dated August 4, 1998, as supplemented by letters dated December 16, 1998, January 12, 1999, and January 28, 1999, PP&L, Inc. (the licensee) submitted a request for changes to the Susquehanna Steam Electric Station (SSES), Unit 2, Technical Specifications (TSs). The requested changes include the Minimum Critical Power Ratio (MCPR) safety limits for ATRIUM™ -10 fuel which is consistent with Siemiens Power Corporation (SPC) 9X9-2 fuel. The SSES-2 Cycle 10 core has 764 fuel assemblies, which consist of 280 fresh ATRIUM™-10 assemblies, 312 once-burned ATRIUM™-10 assemblies, 168 twice-burned 9X9-2 assemblies and 4 twice-burned GE-12 lead use assemblies (LUAs). The December 16, 1998, January 12, 1999, and January 28, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee requested a change to the SSES-2 Cycle 10 TSs in accordance with 10 CFR 50.59, 50.90, and 2.101. The proposed revision of TSs 2.1.1.2, 5.6.5 and the associated Bases 2.1.1.1, 2.1.1.2, and 3.2.2 is described below.

2.1 TS 2.1.1.2

The following changes are proposed for TS 2.1.1.2 Safety Limit MCPR (SLMCPR):

1) Delete Figures 2.1.1.2-1 and 2.1.1.2-2 including the footnote indicating that the MCPR Safety Limit is only approved for Unit 2 Cycle 9, and
2) Single value MCPR Safety Limits of 1.11 for two recirculation loop operation or 1.12 for single recirculation loop operation are proposed to replace the flow dependent MCPR Safety Limit when the reactor steam dome pressure is greater than or equal to 785 psig and core flow is greater than or equal to 10 million lbm/hr.

For the Cycle 10 SLMCPR analyses, the licensee has used the approved ANFB-10 correlation for ATRIUM™ -10 fuel and the ANFB correlation for SPC 9x9-2 fuel to support the TS changes. Based on our review of the licensee's submittals, the staff finds that the proposed TS changes for the SLMCPR of 1.11 for two recirculation loop and 1.12 for single recirculation loop in

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Cycle 10 operation are acceptable. The basis of the staff's acceptance is that boiling transition is predicted using approved critical power correlations and a minimum critical power ratio value is specified such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal or anticipated operational occurrences. The flow dependent MCPR Safety Limits are replaced by single value SLMCPRs using the approved ANFB-10 correlation. Therefore, the deletion of Figures 2.1.1.2-1 and 2.1.1.2-2, which are no longer applicable to SSES-2 Cycle 10 operation, is acceptable.

2.2 TS Bases 2.1.1.1, 2.1.1.2, and 3.2.2

The proposed changes to the Bases are merely to reflect the use of the ANFB-10 correlation for ATRIUM-10 fuel and to update its associated approved methodology in the references. The range of the applicability of the ANFB-10 correlation is valid for pressures greater than 571 psi and bundle mass fluxes greater than 0.115×10^6 lb/hr-ft^2. Therefore, the proposed changes are acceptable.

2.3 TS 5.6.5 Core Operating Limits Report

The staff has reviewed the proposed changes to the list of approved methodologies in TS 5.6.5.b and found them acceptable. These changes revise the list of methodologies to delete references that are no longer directly related to the generation of Core Operating Limits and to add two NRC-approved ANFB-10 topical reports to the list of approved methodologies. These two new topical reports, used in the Cycle 10 specific operating limits, are EMF-1997 (P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation: High Local Peaking Results," July 1998, both listed as number 19, in the list of approved methodologies. The use of these NRC-approved methodologies, which are appropriate for use at SSES-2, will ensure that values for cycle-specific parameters are determined such that all applicable limits of the plant safety analyses are met.

Based on the review, we conclude that the proposed TS revisions are acceptable for SSES-2, Cycle 10 application.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania 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 (63 FR 48262). The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusions set forth in 10 CFR 51.22(c)(9) and (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 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.

Principal Contributor: T. Huang

Date: February 17, 1999