1996

October 11, 1996

Mr. Robert G. Byram
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, PA 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. M95527 AND M95528)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 161 to Facility Operating License No. NPF-14 and Amendment No. 132 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your applications dated May 20 and 28, 1996, as supplemented by letter dated July 25, 1996.

These amendments, for both units, add a reference to the ANF-B critical power correlation to Section 6.9.3.2 of the Technical Specifications (TS), change the values of the minimum critical power ratio (MCPR) in TS Sections 2.1 and 3.4.1.1.2, and make appropriate Bases changes. For Unit 1 only, a reference to ABB licensing methodology report CENPD-300 (for lead use assemblies being used in the reactor core during the upcoming operating cycle) would be added to Section 6.9.3.2.

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

Sincerely,
/S/ J. Stolz for
Chester Poslusny, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-387/388

Enclosures:
1. Amendment No. 161 to License No. NPF-14
2. Amendment No. 132 to License No. NPF-22
3. Safety Evaluation

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cc w/encls: See next page

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

October 11, 1996

Mr. Robert G. Byram
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Pennsylvania Power and Light Company
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These amendments, for both units, add a reference to the ANF-B critical power correlation to Section 6.9.3.2 of the Technical Specifications (TS), change the values of the minimum critical power ratio (MCPR) in TS Sections 2.1 and 3.4.1.1.2, and make appropriate Bases changes. For Unit 1 only, a reference to ABB licensing methodology report CENPD-300 (for lead use assemblies being used in the reactor core during the upcoming operating cycle) would be added to Section 6.9.3.2.

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

Sincerely,

Chester Poslusny, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-387/388

Enclosures:
1. Amendment No. 161 to License No. NPF-14
2. Amendment No. 132 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

Mr. Robert G. Byram
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station,
Units 1 & 2

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. NPF-14

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated May 28, 1996, as supplemented by letter dated July 25, 1996, 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.

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-14 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 161 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

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 11, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 161

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

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
2-1	2-1
B 2-1	B 2-1
B 2-2	B 2-2
3/4 4-1c	3/4 4-1c
B 3/4 4-1	B 3/4 4-1
6-20b	6-20b

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10 million lbm/hr., be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09* with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION:

With MCPR less than 1.09* and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr., be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 AND 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

---

* See Specification 3.4.1.1.2.a for single loop operation requirement.

2.1 SAFETY LIMITS

BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is expected to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specifications 2.1.2 for SPC fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling (ref. ANF-524(P)(A) Revision 2).

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the ANFB correlation is valid for critical power calculations at pressures greater than 585 psig and bundle mass fluxes greater than 0.1×10^6 lbs/hr-ft^2. For operation at low pressures or low flows, the fuel cladding integrity Safety Limit 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 assure a minimum bundle flow for all fuel assemblies which have a relatively high power and potentially can approach a critical heat flux condition. For the SPC 9x9-2 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For the SPC 9x9-2 design, the coolant minimum flow and maximum flow area is such that the mass flux is always greater than 0.25×10^6 lbs/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lbs/hr-ft^2 is 3.35 Mwt or greater. At

2.1.2 THERMAL POWER, High Pressure and High Flow

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR).

The Safety Limit MCPR assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for 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 (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the critical power correlation. ANF-524(P)(A), Revision 2 and PL-NF-90-001-A describe the methodologies used in determining the Safety Limit MCPR.

The ANFB critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlation (refer to Section B 2.1.1), the assumed reactor conditions used in defining the safety limit 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 correlation provide a reasonable degree of assurance that during sustained operation at the Safety Limit MCPR 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 necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission 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 indicates that LWR fuel can survive for an extended period of time in an environment of boiling transition.

SPC fuel is monitored using the ANFB critical power correlation. The effects of channel bow on MCPR are explicitly included in the calculation of the ANFB MCPR Safety Limit. Explicit treatment of channel bow in the ANFB MCPR Safety Limit addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed ≤ 80% of the rated pump speed and the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.10.
2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
≤ 0.58W + 54%	≤ 0.58W + 57%

3. Specification 3.2.2: the APRM Setpoints shall be as follows:

Trip Setpoint	Allowable Value
S ≤ (0.58W + 54%) T	S ≤ (0.58W + 57%) T
S_{RB} ≤ (0.58W + 45%) T	S_{RB} ≤ (0.58W + 48%) T

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the applicable Single Loop Operation MCPR limit as specified in the CORE OPERATING LIMITS REPORT.
5. Specification 3.2.4: The LINEAR HEAT GENERATION RATE (LHGR) shall be less than or equal to the applicable Single Loop Operation LHGR limit as specified in the CORE OPERATING LIMITS REPORT.
6. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ 0.63w + 35%	≤ 0.63W + 37%
	Trip Setpoint	Allowable Value
b. APRM-Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

APPLICABILITY: OPERATIONAL CONDITIONS 1^{*} and 2^{*+}, except during two loop operation.#

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and found acceptable, provided that the unit is operated in accordance with Specification 3.4.1.1.2.

LOCA analyses for two loop operating conditions, which result in Peak Cladding Temperatures (PCTs) below 2200°F, bound single loop operating conditions. Single loop operation LOCA analyses using two-loop MAPLHGR limits result in lower PCTs. Therefore, the use of two-loop MAPLHGR limits during single loop operation assures that the PCT during a LOCA event remains below 2200°F.

The MINIMUM CRITICAL POWER RATIO (MCPR) limits for single loop operation assure that the Safety Limit MCPR is not exceeded for any Anticipated Operational Occurrence (AOO). In addition, the MCPR limits for single-loop operation protect against the effects of the Recirculation Pump Seizure Accident. That is, the radiological consequences of a pump seizure accident from single-loop operating conditions are a small fraction of 10 CFR 100 guidelines.

For single loop operation, the RBM and APRM setpoints are adjusted by a 8.5% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

Surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive reactor vessel internals vibration. Surveillance on differential temperatures below the threshold limits on THERMAL POWER or recirculation loop flow mitigates undue thermal stress on vessel nozzles, recirculation pumps and the vessel bottom head during extended operation in the single loop mode. The threshold limits are those values which will sweep up the cold water from the vessel bottom head.

Specifications have been provided to prevent, detect, and mitigate core thermal hydraulic instability events. These specifications are prescribed in accordance with NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)," dated December 30, 1988.

LPRM upscale alarms are required to detect reactor core thermal hydraulic instability events. The criteria for determining which LPRM upscale alarms are required is based on assignment of these alarms to designated core zones. These core zones consist of the level A, B and C alarms in 4 or 5 adjacent LPRM strings. The number and location of LPRM strings in each zone assure that with 50% or more of the associated LPRM upscale alarms OPERABLE sufficient monitoring capability is available to detect core wide and regional oscillations. Operating plant instability data is used to determine the specific LPRM strings assigned to each zone. The core zones and required LPRM upscale alarms in each zone are specified in appropriate procedures.

CORE OPERATING LIMITS REPORT (Continued)

9. XN-NF-84-97, Revision 0, "LOCA-Seismic Structural Response of an ENC 9x9 Jet Pump Fuel Assembly," Exxon Nuclear Company, Inc., December 1984.

10. PLA-2728, "Response to NRC Question : Seismic/LOCA Analysis of U2C2 Reload," Letter from H.W. Keiser (PP&L) to E. Adensam (NRC), September 25, 1986.

11. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 Fuel," May 1988.

12. 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.

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

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

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

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

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

18. 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," September 1994.

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

20. CENPD-300-P, "Reference Safety Report for Boiling Water Reactor Reload Fuel", ABB Combustion Engineering Nuclear Operations, November 1994.

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

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

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.132
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 the Pennsylvania Power & Light Company, dated May 20, 1996, as supplemented by letter dated July 25, 1996, 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.

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) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 132 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

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 11, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 132

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
2-1	2-1
B 2-1	B 2-1
B 2-2	B 2-2
3/4 4-1c	3/4 4-1c
B 3/4 4-1	B 3/4 4-1
6-20a	6-20a
6-20b	6-20b

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10 million lbm/hr., be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08* with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION:

With MCPR less than 1.08* and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr., be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 AND 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

---

* See Specification 3.4.1.1.2.a for single loop operation requirement.

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is expected to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specification 2.1.2 for SPC fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling (ref. ANF-524(P)(A) Revision 2).

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the ANFB correlation is valid for critical power calculations at pressures greater than 585 psig and bundle mass fluxes greater than 0.1×10^6 lbs/hr-ft^2. For operation at low pressures or low flows, the fuel cladding integrity Safety Limit 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 assure a minimum bundle flow for all fuel assemblies which have a relatively high power and potentially can approach a critical heat flux condition. For the SPC 9x9-2 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For the SPC 9x9-2 design, the coolant minimum flow and maximum flow area is such that the mass flux is always greater than 0.25×10^6 lbs/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lbs/hr-ft^2 is 3.35 Mwt or greater. At 25% thermal power a bundle power of 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% of RATED THERMAL POWER for reactor pressures below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR).

The Safety Limit MCPR assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for 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 (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. ANF-524 (P)(A), Revision 2 and PL-NF-90-001(A) and Supplement 2 describe the methodologies used in determining the Safety Limit MCPR.

The ANFB critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlation (refer to Section B 2.1.1), the assumed reactor conditions used in defining the safety limit 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 correlation provide a reasonable degree of assurance that during sustained operation at the Safety Limit MCPR 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 necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission 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 indicates that LWR fuel can survive for an extended period of time in an environment of boiling transition.

SPC fuel is monitored using the ANFB Critical Power Correlation. The effects of channel bow on MCPR are explicitly included in the calculation of the ANFB MCPR Safety Limit. Explicit treatment of channel bow in the ANFB MCPR Safety Limit addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

REACTOR COOLANT SYSTEM

RECIRCULATION LOOPS-SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed ≤ 80% of the rated pump speed and the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.09.
2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
≤ 0.58W + 54%	≤ 0.58W + 57%

3. Specification 3.2.2: the APRM Setpoints shall be as follows:

Trip Setpoint	Allowable Value
S ≤ (0.58W + 54%) T	S ≤ (0.58W + 57%) T
S_{RB} ≤ (0.58W + 45%) T	S_{RB} ≤ (0.58W + 48%) T

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the applicable Single Loop Operation MCPR limit as specified in the CORE OPERATING LIMITS REPORT.
5. Specification 3.2.4: The LINEAR HEAT GENERATION RATE (LHGR) shall be less than or equal to the applicable Single Loop Operation LHGR limit as specified in the CORE OPERATING LIMITS REPORT.
6. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ 0.63w + 35%	≤ 0.63W + 37%
	Trip Setpoint	Allowable Value
b. APRM-Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*+, except during two loop operation.#

ACTION:

a. In OPERATIONAL CONDITION 1:
   1. With
      a) no reactor coolant system recirculation loops in operation, or
      b) Region I of Figure 3.4.1.1.1-1 entered, or
      c) Region II of Figure 3.4.1.1.1-1 entered and core thermal hydraulic instability occurring as evidenced by:

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and found acceptable, provided that the unit is operated in accordance with Specification 3.4.1.1.2.

LOCA analyses for two loop operating conditions, which result in Peak Cladding Temperatures (PCTs) below 2200°F, bound single loop operating conditions. Single loop operation LOCA analyses using two-loop MAPLHGR limits result in lower PCTs. Therefore, the use of two-loop MAPLHGR limits during single loop operation assures that the PCT during a LOCA event remains below 2200°F.

The MINIMUM CRITICAL POWER RATIO (MCPR) limits for single loop operation assure that the Safety Limit MCPR is not exceeded for any Anticipated Operational Occurrence (AOO). In addition, the MCPR limits for single-loop operation protect against the effects of the Recirculation Pump Seizure Accident. That is, the radiological consequences of a pump seizure accident from single-loop operating conditions are a small fraction of 10CFR100 guidelines.

For single loop operation, the RBM and APRM setpoints are adjusted by a 8.5% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

Surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive reactor vessel internals vibration. Surveillance on differential temperatures below the threshold limits of THERMAL POWER or recirculation loop flow mitigates undue thermal stress on vessel nozzles, recirculation pumps and the vessel bottom head during extended operation in the single loop mode. The threshold limits are those values which will sweep up the cold water from the vessel bottom head.

Specifications have been provided to prevent, detect, and mitigate core thermal hydraulic instability events. These specifications are prescribed in accordance with NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)," dated December 30, 1988.

LPRM upscale alarms are required to detect reactor core thermal hydraulic instability events. The criteria for determining which LPRM upscale alarms are required is based on assignment of these alarms to designated core zones. These core zones consist of the level A, B and C alarms in 4 or 5 adjacent LPRM strings. The number and location of LPRM strings in each zone assure that with 50% or more of the associated LPRM upscale alarms OPERABLE sufficient monitoring capability is available to detect core wide and regional oscillations. Operating plant instability data is used to determine the specific LPRM strings assigned to each zone. The core zones and required LPRM upscale alarms in each zone are specified in appropriate procedures.

An inoperable jet pump is not in itself, a sufficient reason to declare a recirculation loop inoperable, but it does in case of a design basis accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

CORE OPERATING LIMITS REPORT (Continued)

6.9.3.2 The analytical methods used to determine the core operating limits shall be those topical reports and those revisions and/or supplements of the topical report previously reviewed and approved by the NRC, which describe the methodology applicable to the current cycle. For Susquehanna SES the topical reports are:

1. PL-NF-87-001-A, "Qualification of Steady State Core Physics Methods for BWR Design and Analysis," July, 1988.

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

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

4. 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.

5. 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.

6. PLA-3407, "Proposed Amendment 132 to License No. NPF-14: Unit 1 Cycle 6 Reload, "Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), July 2, 1990.

7. Letter from Elinor G. Adensam (NRC) to H. W. Keiser (PP&L), "Issuance of Amendment No. 31 to Facility Operating License No. NPF-22 - Susquehanna Steam Electric Station, Unit 2," October 3, 1986.

8. PLA-3533, Revised Proposed Amendment 67 to License No. NPF-22: Unit 2 Cycle 5 Reload, "Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), March 7, 1991.

9. XN-NF-84-97, Revision 0, "LOCA-Seismic Structural Response of an ENC 9x9 Jet Pump Fuel Assembly," Exxon Nuclear Company, Inc., December 1984.

10. PLA-2728, "Response to NRC Question: Seismic/LOCA Analysis of U2C2 Reload," Letter from H.W. Keiser (PP&L) to E. Adensam (NRC), September 25, 1986.

11. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 Fuel," May 1988.

12. 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.

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

CORE OPERATING LIMITS REPORT (Continued)

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

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

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

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

18. 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," September 1994.

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

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

21. PL-NF-90-001, Supplement 2, "Application of Reactor Analysis Methods to BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation."

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

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.161 TO FACILITY OPERATING LICENSE NO. NPF-14

AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 388

1.0 INTRODUCTION

By letters dated May 20 and 28, 1996, as supplemented by letter dated July 25, 1996, the Pennsylvania Power and Light Company (PP&L or the licensee) submitted a request for changes to the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Technical Specifications (TSs). The requested changes for Units 1 and 2 would: add a reference to the ANF-B critical power correlation to Section 6.9.3.2 of the TSs; change the values of the minimum critical power ratio (MCPR) in TS Sections 2.1 and 3.4.1.1.2; and make appropriate Bases changes. For Unit 1 only, a reference to ABB licensing methodology report CENPD-300 (for lead use assemblies being used in the reactor core during the upcoming operating cycle) would be added to Section 6.9.3.2.

2.0 EVALUATION

By letter dated May 29, 1996, the staff approved Topical Report PL-NF-90-001, Supplement 2, "Application of Reactor Analysis Methods for BWR Design and Analysis," which had been submitted by PP&L on August 1, 1995. The letter approved two changes to PP&L's current licensing methodology. The first change approved replacement of the CPM-2 lattice physics code with the CASMO-36 code. The second change approved was to replace the XN-3 critical power correlation with the ANF-B correlation. The ANF-B correlation was developed to provide a generic tool for evaluation of the departure from nucleate boiling heat flux for all ANF BWR fuel designs. The staff had previously approved the ANF-B correlation on a generic basis.

In the May 29, 1996 letter, the staff advised PP&L that the changes to the proposed rotated bundle analysis would be reviewed separately, since this issue was still under generic review by the staff. In the letter, however, PP&L was informed that they could continue to use the currently approved rotated bundle analysis methodology and could incorporate the ANF-B critical power correlation into the rotated bundle analysis.

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2.1 ANF-B Correlation (TS Sections 2.1 and 3.4.1.1.2)

The current request for a change to the TS reflects the application of the staff-approved ANF-B methodology to SSES reload analyses and specifically the calculation of the MCPR safety limits. In the licensee's submittals dated May 20 and 28, 1996, the licensee stated that its contractor, Siemens Power Corporation (SPC), performed a cycle-specific MCPR safety limit analysis for both Units 1 and 2. The licensee further stated in these submittals:

> This MCPR calculation statistically combines uncertainties on feedwater flow, feedwater temperature, core flow, core pressure, core power distribution, and the uncertainty in the Critical Power Correlation. The SPC analysis uses cycle specific power distributions and calculates a Safety in the critical power correlation. The SPC analysis uses cycle specific power distributions and calculates a Safety Limit MCPR such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences.

The proposed amendment would increase the TS 2.1.2 MCPR Safety Limit from 1.06 to 1.09 for Unit 1 (1.10 for single loop operation) and from 1.06 to 1.08 for Unit 2 (1.09 for single loop operation). The proposed TS changes in these submittals reflect the use of a staff-approved method, include the calculational results and add this information and references in the appropriate TS sections and the Bases. The staff therefore finds these changes acceptable as they will ensure appropriate design margins and operating limits are maintained, thus ensuring core integrity.

2.2 Changes Related to ABB Lead Use Assemblies (TS Section 6.9.3.2)

As indicated in the licensee's submittal, PP&L is inserting four ABB lead use fuel assemblies in the Unit 1 Cycle 10 core, which are of a different mechanical design from the current Siemens 9x9-2 fuel used in the unit. Accordingly, the licensee plans to use the ABB methodology, described in CENPD-300-P, "Reference Safety Report for Boiling Water Reactor Reload Fuel," dated November 1994, to calculate the operating limits for these four assemblies. This topical report has already been approved by the staff for use by plants who utilize this type of fuel and the analysis is applicable to SSES and the results are acceptable. The submittal proposes the addition of a reference to this ABB topical report in Section 6.9.3.2. The methodology described in CENPD-300-P will be used to calculate the operating limits for the four Lead Use Assemblies for the Core Operating Limits Report (COLR). The licensee is also proposing the other following changes to the list of approved references for the COLR in Section 6.9.3.2:

a. For present reference 13 (an Exxon topical), substitute: ANF-524(P)(A), Revision 2 and Supplement 1, Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," November 1990.

b. For the present reference 14 (the XN-3 Critical Power Correlation), substitute: ANF-1125(P)(A) and ANF-1125(P)(A), Supplement 1, "ANFB Critical Power Correlation," April 1990.

c. The CENPD-300-P topical report discussed above is being added as a new reference 20. However, the correct designation for this topical report should include an "(A)" since it was approved by the NRC's letter of May 29, 1996. (PP&L's submittal was dated May 28, 1996, prior to the NRC's letter of approval.)

d. A new reference, PL-NF-90-001, Supplement 2 is being added as Reference 21. This topical report was approved by NRC's letter of May 29, 1996.

The staff finds this change acceptable because the use of NRC-approved methodology will ensure that values for cycle-specific parameters are determined consistent with applicable limits of the plant safety analysis. All of the four above topical reports have been previously approved for use by PP&L as acceptable methodology in analyzing core parameters for the Siemens fuel used in Susquehanna, Units 1 and 2. (All fuel is Siemens fuel, except for 4 GE LUAs in Unit 1 and the proposed 4 LUAs that will be in Unit 1).

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania 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. 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 (61 FR 44362 and 61 FR 47529). Accordingly, the amendments meet 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.

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.

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Date:October 11, 1996