May 7, 1997

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, UNIT 2 (TAC NO. M97499)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 139 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 December 18, 1996, as supplemented on February 26, March 12 and 27, April 3, 9, 16, 18, and 24, 1997.

This amendment authorizes the use of ATRIUM-10 fuel in Unit 2 beginning with Cycle 9 under all operational Conditions as defined in the revised TSs.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be published in the Federal Register.

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

Docket No. 50-388

Enclosures: 1. Amendment No. 139 to License No. NPF-22
2. Safety Evaluation

cc w/encls: See next page

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

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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. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
License No. NPF-22

I. 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 December 18, 1996 as supplemented on February 26, March 12 and 27, April 3, 9, 16, 18, and 24, 1997, 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) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 139, 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 upon receipt by the licensee.

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: May 7, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 139

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
xxii	xxii
1-1	1-1
2-1	2-1
---	2-2a
B 2-1	B 2-1
B 2-2	B 2-2
3/4 4-1c	3/4 4-1c
---	3/4 4-1g
5-6	5-6
6-20b	6-20b
---	6-20c

INDEX

LIST OF FIGURES

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE EXPOSURE

1.2 The AVERAGE BUNDLE EXPOSURE shall be equal to the total energy produced by the bundle divided by the total initial weight of uranium in the fuel bundle.

The AVERAGE PLANAR EXPOSURE at a specified height shall be equal to the total energy produced per unit length at the specified height divided by the total initial weight of uranium per unit length at that height.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at the height.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

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 the value shown in Figure 2.1.2-1*# 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 the value shown in Figure 2.1.2-1*# 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.

\# Only applicable for Unit 2 Cycle 9 operation.

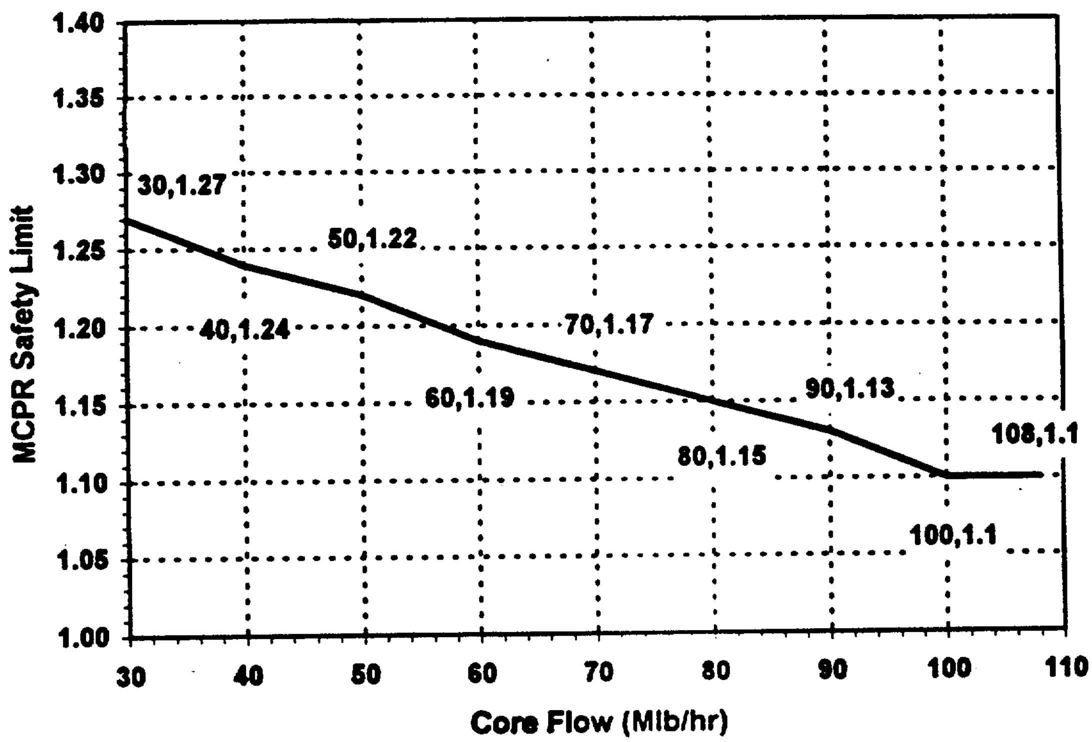

Figure 2.1.2-1
MCPR Safety Limit vs Core Flow

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 ATRIUM-10 design, the minimum bundle flow is greater than 28,000 lbs/hr. For both the SPC 9x9-2 and ATRIUM-10 fuel designs, 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 ATLAS test data taken at pressures from 14.7 psia to 800 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 critical power correlation. Section 6.9.3.2 contains the methodologies used in determining the Safety Limit MCPR.

The ANFB critical power correlation is 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), 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."

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 the value shown in Figure 3.4.1.1.2-1[++].
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:

++ Only applicable for Unit 2 Cycle 9 operation.

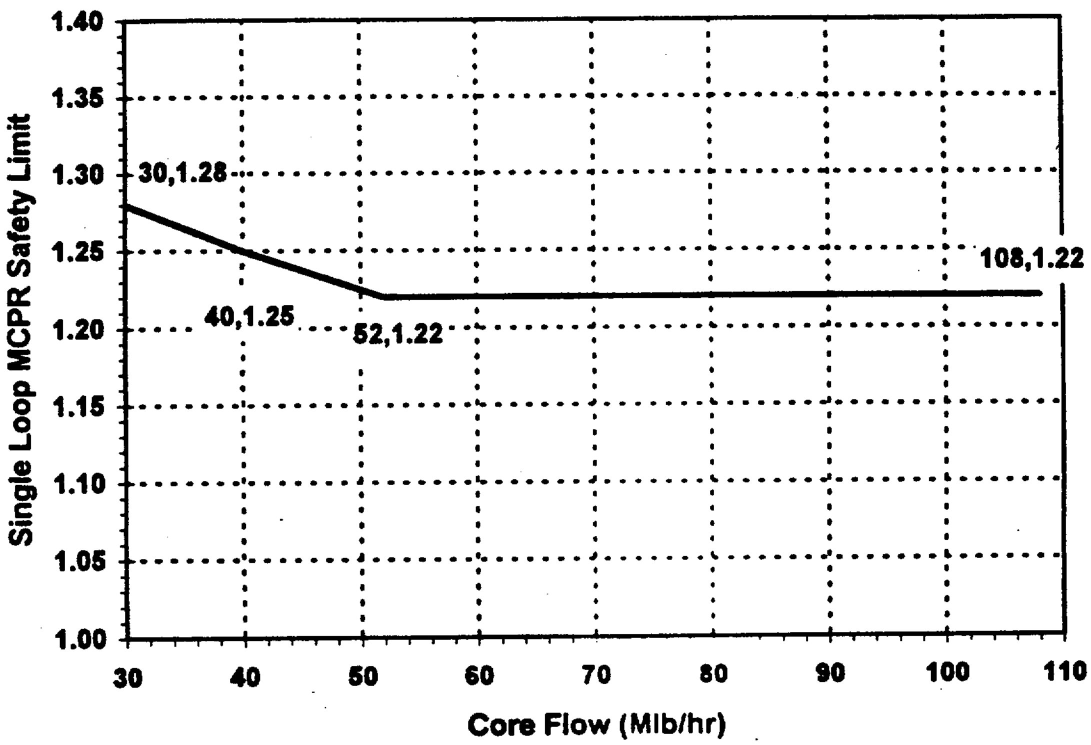

Figure 3.4.1.1.2-1
MCPR Safety Limit vs Core Flow
Single Loop Operation

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of non-enriched or slightly enriched uranium dioxide as fuel material and water rods or water channels. Limited substitutions of Zirconium alloy filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or analyses to comply with all fuel safety design bases. A limited number of lead use assemblies that have not completed representative testing may be placed in non-limiting core regions. Reload fuel shall have a maximum lattice average enrichment of 4.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C), and/or Hafnium metal. The control rod shall have a nominal axial absorber length of 143 inches. Control rod assemblies shall be limited to those control rod designs approved by the NRC for use in BWRs.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.

2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal T_{ave} of 532°F.

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-A, "Application of Reactor Analysis Methods to BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation."

22. 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.

23. XN-NF-81-58(P)(A) Supplements 1 and 2 Revision 2, "RODEX 2 Fuel Rod Thermal-Mechanical Response Evaluation Model," May 1986.

24. XN-NF-85-74(P)(A), "RODEX 2A (BWR) Fuel Rod Thermal-Mechanical Response Evaluation Model," August 1986.

25. XN-NF-82-06(P)(A) and Supplements 2, 4, and 5 Revision 1, "Qualification of Exxon Nuclear Fuel for Extended Burnup," October 1986.

26. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity," November 1986.

27. ANF-90-082(P)(A) Revision 1 and Revision 1 Supplement 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," May 1995.

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

29. ANF-CC-33(P)(A) Supplement 2, "HUXY : A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," January 1991.

30. XN-CC-33(P)(A) Revision 1, "HUXY : A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option Users Manual," November 1975.

31. 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.

32. 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.

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

34. ANF-1358(P)(A), Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," September 1992.

35. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2 : A Computer Program for Boiling Water Reactor Transient Analyses," August 1990.

36. 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.

37. XN-NF-84-105(P)(A), Volume 1 Supplement 4, "XCOBRA-T : A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, Void Fraction Model Comparison to Experimental Data," June 1988.

38.# EMF-97-010, Revision 1, "Application of ANFB to ATRIUM™-10 for Susquehanna Reloads," March 1997.

39.# PLA-4595, "Response to NRC Request For Additional Information On Siemens' Report EMF-97-010, Revision 1," March 27, 1997.

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.

Only applicable for Unit 2 Cycle 9 operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.139TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-388

1.0 INTRODUCTION

By letter dated December 18, 1996 (Reference 1, PLA-4527), as supplemented by letters dated February 26, 1997 (Reference 2, PLA-4572), March 12, (Reference 3, PLA 4582), March 27, 1997 (Reference 4, PLA-4595), April 3, 1997 (Reference 5, PLA-4599), April 9, 1997 (Reference 6, PLA-4605), April 16, 1997 (Reference 7, PLA-4611), and April 18, 1997 (Reference 8, PLA-4613), and April 24, 1997 (Reference 9, PLA-4620), Pennsylvania Power & Light Company (PP&L, the licensee) proposed changes to the Technical Specifications (TSs) for the Susquehanna Steam Electric Station, Unit 2 Cycle 9 (S2C9) which is the first 24-month operating cycle. The requested changes would authorize the use of ATRIUM-10 fuel in Unit 2 beginning with cycle 9 under all operational Conditions (1-5) as defined in the TSs. The proposed changes include the Safety Limits Minimum Critical Power Ratio (SLMCPR) based on the cycle-specific analysis of the mixed core of Siemens Power Corporation (SPC) ATRIUM-10 and SPC 9x9-2 fuel parameters and other sections of the TSs relating to the use of ATRIUM-10 fuel. Due to the limitations imposed in the approved Advanced Nuclear Fuel-B (ANFB) Critical Power Correlation (ANF-1125 (P) (A) and its Supplements 1 and 2) and the findings in the inspection of the Application of ANFB to ATRIUM-10 for Susquehanna Reload at Siemens Power is Corporation (SPC) in February 1997, this review based on the updated information provided in References 2 through 9, and its findings relative to the minimum critical power ratio (MCPR) limits and the use of two new methodologies are applicable only to the ninth Susquehanna Unit 2 reload (S2C9).

During the staff's review of the TS changes discussed in this safety evaluation, the licensee made two exigent amendment requests. The first requested TS change was made to permit the loading of the Atrium-10 fuel into the core and maintaining the reactor in Condition 5, refueling. This amendment (#136) was approved on April 9, 1997. The second requested TS change was made to permit the reactor to be brought into Condition 4 (cold shutdown) and Condition 3 (hot shutdown) to permit certain testing to be conducted. This amendment (#138) was approved on April 25, 1997. Both of these TS changes authorized by the amendments noted above are being modified by the current TS revisions to enable the fuel to be used under all operational conditions and to include applicable references and safety limits.

Brookhaven National Laboratory (BNL) assisted the NRC staff in the review of EMF-97-010 (P), Revision 1 (provided to the Commission as an attachment to Reference 4) and prepared a technical evaluation report (TER) which is attached to this safety evaluation (SE) to support the review for the SLMCPR TS changes.

2.0 <u>EVALUATION</u>

2.1 Mechanical Design

The ATRIUM-10 fuel design is a 10x10 lattice design which contains 83 full length fuel rods, 8 part length fuel rods, and a central water channel to enhance neutron moderation. The ATRIUM-10 fuel design was analyzed and assessed by Siemens according to the approved methodology, entitled "Generic Mechanical Design Criteria for BWR Fuel Designs," ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1. The staff has performed an on-site audit of ATRIUM-10 fuel at Siemens. Although the staff discovered a procedural deficiency, we conclude that, with the correction of the deficiency, the ATRIUM-10 fuel mechanical design followed the approved methodology, and therefore, is acceptable for Susquehanna 2 Cycle 9.

2.2 Application of the ANFB Critical Power Correlation to ATRIUM-10 Fuel

The review of the Siemens reload analysis for Cycle-9 of Susquehanna-2 and the application of the ANFB correlation to the ATRIUM-10 fuel design was included in the NRC Vendor Inspection (No. 99900081/97-01) at the Siemens Power Corporation Facility in Richland, WA, during the week of February 9-14, 1997. Several important concerns were identified during this review of the SSE-2 reload analysis and the application of ANFB to the ATRIUM-10 fuel design. First, it was noted that the local fuel rod power peaking for SSE-2 Cycle-9 fuel bundles exceeded the range of the ANFB correlation as stated in Reference 4 (local peaking < 1.3). In addition, a flow dependent bias in the ANFB correlation was identified which resulted in the nonconservative overprediction of the measured critical power at low flows. Both of these effects were outside the presently approved applicable SPC methodologies.

In response to findings of the NRC vendor inspection at the SPC during the week of February 9-14, 1997, PP&L has submitted a revised ANFB methodology and core flow dependent MCPR safety limits and a supporting topical report, EMF-97-010(P), Revision 1, "Application of ANFB to ATRIUM-10 for Susquehanna Reloads," March 1997 (Reference 3) for S2C9 reload. The detailed review was given in the attached TER (Attachment), which was provided by our consultant at BNL. The staff adopts the findings and position included in this report.

Accordingly, the staff concludes that the revised methodology and newly revised core flow dependent MCPR safety limits, as proposed in the TS change, are acceptable for the SSES Unit 2 S2C9 reload. The revisions to the TS address the staff's concerns about local fuel rod power peaking and fuel behavior at low reactor coolant recirculation flow (flow bias).

2.3 Technical Specification Changes

The licensee requested a change to the S2C9 TSs in accordance with 10 CFR 50.90. The proposed revisions of the TS and its associated Bases - are described below.

(1) TS 1.2 Average Exposure and 1.3 Average Planar Linear Heat Generation Rate

The proposed changes of definitions for TS 1.2 average bundle exposure and average planar exposure and for TS 1.3 average planar linear heat generation rate to reflect the use of part length rods in the ATRIUM-10 fuel assemblies as well as full length rods in SPC 9x9-2 fuel assemblies are acceptable since the new wordings clearly define the meaning for the new fuel assemblies used. This change is not restricted to S2C9.

(2) TS 2.1.2 and 3.4.1.1.2 and Bases 2.1.1 and 2.1.2

The safety limit MCPR in TS 2.1 and its associated Bases 2.0 is proposed to change from 1.08 to the value shown in Figure 2.1.2-1 ($\geq$ 1.11 depending on the core flow) for operation with two recirculation loops with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr., and from 1.09 to the value shown in Figure 3.4.1.1.2-1 ($\geq$1.22) for single loop operation (SLO) based on the cycle-specific analysis of a modified ANFB core flow dependent SLMCPR performed by SPC for S2C9 mixed core of ATRIUM-10/SPC 9x9-2 fuel (Reference 8).

The staff in conjunction with our consultant at BNL has reviewed the proposed TS and its associated Bases changes and has found them acceptable since they are based on the analyses performed using S2C9 cycle-specific inputs and approved methodologies in Reference 3. The details of our evaluation are provided in the attachment to this safety evaluation.

The staff noted that in the submittal dated April 3, 1997, the licensee added a footnote for TS Section 3.4.1.1.2 designated with the "#" symbol. In recent discussions, the licensee discovered that this footnote symbol had already been used in that TS section. Accordingly, the licensee changed the applicable footnote symbol to "++" which is an administrative change and found to be acceptable by the staff.

(3) TS 5.3.1 - Fuel Assemblies

Section 5.3.1 was revised to reflect the use of ATRIUM-10 fuel with a central water channel, part length fuel rods and different active fuel length from that of SPC 9x9-2. The maximum allowed enrichment was increased from 4.0 to 4.5 weight percent U-235 which is consistent with 10 CFR 51.52. The revised Section will be read as follows.

> "...or slightly enriched uranium dioxide as fuel material and water rods or water channels. ... Reload fuel shall have a maximum lattice average enrichment of 4.5 weight percent U-235."

The ATRIUM-10 fuel design increases the maximum enrichment from 4.0 to 4.5 weight percent U-235 and allows a 24-month operating cycle. The enrichment change was approved, by Amendment No. 136, April 9, 1997, however, dose consequences of this change were not considered at that time because the plant was not permitted to startup or become critical. The radiological consequences of design basis accidents will not be affected by the enrichment change after the fuel is used under all operational conditions, except as discussed below. The licensee in its April 24, 1997 submittal indicated that the maximum discharge exposure for the ATRIUM-10 fuel is 48 MWd/kgU (MWD/MTU) as documented in the SPC report EMF-95-52(P), "Mechanical Design Evaluation for Siemens Power Corporation ATRIUM-10 BWR Reload Fuel, dated July, 1995. This burnup rate is greater than the 45 MWD/MTU value evaluated in the Commission Safety Evaluation dated September 12, 1995.

In addition to the information provided in the submittals by the licensee, the staff has reviewed a publication which was prepared for the NRC entitled, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," NUREG/CR 5009, February 1988. The NRC contractor, Pacific Northwest Laboratory (PNL) of Batelle Memorial Institute, examined the changes that could result in the NRC design basis accident (DBA) assumptions, described in the various appropriate Standard Review Plan (SRP) sections and/or Regulatory Guides (RG), that could result from the use of extended burnup fuel (up to 60 MWD/MTU). The staff finds that the only DBA that could be affected by the use of extended burnup fuel, even in a minor way, would be the potential thyroid doses that could result from a fuel handling accident (with fuel that had been subject to the maximum burnup). PNL estimated that I-131 fuel gap activity in the peak fuel rod with 60 MWD/MTU burnup could be as high as 12%. This value is approximately 20% higher than the value normally used by the staff in evaluating the fuel handling accidents (as per RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors).

For the fuel handling accident, PNL concluded that the use of RG 1.25 procedures for the calculation of accident doses for extended burnup fuel may be utilized. These procedures give conservative estimates for noble gas release fractions that are above calculated values for peak rod burnups of 60 MWD/MTU. Iodine-131 inventory, however, may be up to 20% higher than that predicted by RG 1.25 procedures.

In its evaluation for the Susquehanna units issued in April 1981 (NUREG-0776), the staff conservatively estimated offsite doses due to radionuclides released to the atmosphere from a fuel handling accident. The staff concluded that the plant mitigative features would reduce the doses for this DBA to below the doses specified in the SRP Section 15.7.4. In the Safety Evaluation dated September 12, 1995, the staff reanalyzed the fuel handling accident based on a maximum fuel burnup of 45 MWD/MTU using the information from PNL discussed above. Table 1 below was included in that Safety Evaluation. The evaluation presented in the Table continues to bound the licensee's current proposal to use fuel with 4.5 weight percent U-235 with a maximum burnup of 48 MWD/MTU.

Table 1

Radiological Consequences of Fuel Handling Design Basis Accident (rem)

	Thyroid	
	Exclusion Area	Low Population Zone
Staff Evaluation April 1981 (NUREG-0776)	12	<1
Bounding Estimates For Extended Burnup Fuel 5% Enrichment	14.4	<1.2
Dose Acceptance Criterion (NUREG-0800 Section 15.7.4)	75	75

The staff therefore concludes that the only potential increased doses resulting from the fuel handling accidents with extended burnup fuel with increased U-235 enrichment are the thyroid doses; these doses remain well within the dose limits given in NUREG-0800 and are therefore acceptable.

Based on the staff evaluation, we conclude that this revision is consistent with the staff position and thus acceptable for Susquehanna 2. This approval is not restricted to S2C9.

(4) TS 6.9.3.2 - Core Operating Limits Report

The proposed change is to add additional approved methodologies relating to the use of SPC ATRIUM-10 fuel assemblies. The proposed approved methodologies are the following:

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

(b) XN-NF-81-58(P)(A) Supplements 1 and 2, "RODEX 2 Fuel Rod Thermal-Mechanical Response Evaluation Model," May 1986.

(c) XN-NF-85-74(P)(A), "RODEX 2A (BWR) Fuel Rod Thermal-Mechanical Response Evaluation Model," August 1986.

(d) XN-NF-82-06(P)(A) and Supplements 2, 4, and 5 Revision 1, "Qualification of Exxon Nuclear Fuel for Extended Burnup," October 1986.

(e) XN-NF-85-92(P)(A), "EXXON Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity," November 1986.

(f) ANF-90-082(P)(A) Revision 1 and Revision 1 Supplement 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," May 1995.

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

(h) ANF-CC-33(P)(A) Supplement 2, "HUXY : A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," January 1991.

(i) ANF-CC-33(P)(A) Revision 1, "HUXY : A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option Users Manual," November 1975.

(j) 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.

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

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

(m) ANF-1358(P)(A) Volume 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," September 1992.

(n) ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," August 1990.

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

(p) XN-NF-84-105(P)(A), Volume 1 Supplement 4, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, Void Fraction Model Comparison to Experimental Data," June 1988.

(q)* EMF-97-010(P), Revision 1, "Application of ANFB to ATRIUM-10 for Susquehanna Reloads," March 1997.

(r)* PLA-4595, "Response to NRC Request for Additional Information on Siemens' Report EMF-97-010, Revision 1, March 27, 1997.

"*" Only Applicable for S2C9 Operation.

The staff has concluded that the generic methodologies (a)-(p) are applicable to this plant-specific ATRIUM-10 fuel design. Based on our review, we also conclude that methodologies (q) and (r) are acceptable for only S2C9 application since the proposed fuel design has been analyzed on a plant and cycle-specific basis using the NRC approved methodologies. This application of the methodologies resolves a previous staff concern about the ANF-B correlation discussed in Amendment No. 136, dated April 9, 1997, and enables the plant to proceed to Conditions 2 (startup) and 1 (operation). Finally, the staff concludes that the licensee may use this fuel under all operational conditions for S2C9 since 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 limits have been met.

3.0 PUBLIC COMMENTS

No public comments were received from the notice of this amendment dated March 12, 1997 and published in the Federal Register on March 18, 1997 (62 FR 12859). However, some comments were received from the public in response to a notice, published in two local newspapers, the Berwick Press Enterprise, Berwick, PA, and the Wilkes Barre Times Leader, Wilkes Barre, PA, April 22-24, 1997 concerning the exigent amendment that would enable the plant to move to Conditions 4 and 3 using the Atrium-10 fuel. The staff considered the following comments applicable to this amendment.

One comment from an individual was a request that the NRC completely review the new fuel design to ensure that it is as safe as the current fuel. As discussed above, the NRC technical staff with assistance from Brookhaven National Laboratory has conducted an audit at Siemens Power Corporation and has completed a comprehensive review of this new fuel design and analyses which support the safe operation of the reactor. Issues raised by the staff have resulted in the licensee proposing as noted in the revised TS to operate the fuel with conservative safety limits that provide an additional level of safety in the manner in which the fuel will be utilized to produce heat in the reactor core especially when the reactor coolant flow is at low levels. The staff has no reason to believe that the new fuel will not be as safe as the fuel used in the reactor up to this point in time.

Another individual voiced opposition to the use of the new fuel unless there was assurance that there would be no increase in risk to the public given an accident. Another comment was a concern that the use of the new fuel could potentially result in more radiation being released after an accident than compared to that which could be released by an accident with the current reactor fuel loaded in the core. The staff considered the fact that the new fuel reflected an enrichment increase from 4.0% to 4.5% uranium-235 and as discussed in the safety evaluation considered the maximum burnup rate provided by the licensee and the limiting accident that could produce the maximum dose to the public. Given these facts, the staff still determined that the consequences would still be well below 10 CFR Part 100 release limits. This TS change and use of the ATRIUM-10 fuel was found not to result in a significant increase in the consequences of an accident previously analyzed for the plant and therefore is acceptable to the staff.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on May 6, 1997 (62 FR 24669). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: Technical Evaluation Report
No. PLA-4527 by BNL dated
March 27, 1997

Principal Contributor: T. Huang

Date: May 7, 1997

7.0 REFERENCES

1. PLA-4527, Susquehanna Steam Electric Station Proposed Amendment No. 166 to License NPF-22: Unit 2 Technical Specification Changes for ATRIUM-10 Fuel, December 18, 1996.

2. PLA-4572, Susquehanna Steam Electric Station Correction to Proposed Amendment No. 166 to License NPF-22: Unit 2 Technical Specification Changes for ATRIUM-10 Fuel, February 26, 1997.

3. PLA-4582, Susquehanna Steam Electric Station Addendum to proposed Amendment No. 166 to License NPF-22: Revised ANFB Methodology and Core Flow Dependent MCPR Safety Limits, March 12, 1997.

4. PLA-4595, Susquehanna Steam Electric Station Response to NRC Request for Additional Information on SIEMENS' Report EMF-97-010, Rev. 1, March 27, 1997.

5. PLA-4599, Susquehanna Steam Electric Station Addendum #2 to Proposed Amendment No. 166 to License NPF-22: Addition of Limiting Footnotes and a Reference Reflecting PP&L's RAI Response, April 3, 1997.

6. PLA-4605, Susquehanna Steam Electric Station Response to NRC Request for Additional Information on PP&L's Proposed Amendment No. 166 to License No. NPF-22: Unit 2 Technical Specification Changes for ATRIUM-10 Fuel, April 9, 1997.

7. PLA-4611, Susquehanna Steam Electric Station Addendum to PP&L's Response to NRC Request for Additional Information on PP&L's Proposed Amendment No. 166 to License No. NPF-22: Unit 2 Technical Specification Changes for ATRIUM-10 Fuel, April 16, 1997.

8. PLA-4613, Susquehanna Steam Electric Station Addendum #3 to Proposed Amendment No. 166 to License NPF-22: Revised Core Flow Dependent MCPR Safety Limits, April 18, 1997.

9. PLA-4620, Susquehanna Steam Electric Station Response to NRC Question on Proposed Amendment No. 166 to License NPF-22: ATRIUM-10 Exposure Limit, April 24, 1997.

TECHNICAL EVALUATION REPORT

Report Title: Susquehanna Steam Electric Station Proposed Amendment No. 166 to License NPF-22: Unit-2 Technical Specification Changes for ATRIUM™-10 Fuel

Report Number: PLA-4527

Report Date: March 27, 1997

Docket No.: 50-388

Originating Organization: Pennsylvania Power & Light Company

1.0 INTRODUCTION

The Pennsylvania Power & Light Company (PP&L) has submitted in Reference 1 the proposed changes to the Susquehanna Steam Electric Unit-2 (SSE-2) Technical Specifications for NRC review and approval. These Technical Specification changes result primarily from the use of the new Siemens Power Corporation (SPC) ATRIUM™-10 fuel. Specifically, these changes involve the application of the Siemens ANFB critical power correlation to the ATRIUM™-10 fuel design in determining the Operating Limit MCPR, and are based on the Siemens Topical Report EMF-97-010 (Reference 2). EMF-97-10 provides test data taken specifically to support the application of the ANFB correlation to the ATRIUM™-10 fuel design and the determination of the correlation additive constants. The change in the ANFB correlation additive constants required for the ATRIUM™-10 fuel design affects both the Operating Limit MCPR (OLMCPR) and Safety Limit MCPR (SLMCPR).

The review of the Siemens reload analysis for Cycle-9 of Susquehanna-2 and the application of the ANFB correlation to the ATRIUM™-10 fuel design was included in the NRC Vendor Inspection (No. 99900081/97-01) at the Siemens Power Corporation Facility in Richland, WA during the week of February 9-14, 1997. Several important concerns were identified during this review of the SSE-2 reload analysis and the application of ANFB to the ATRIUM™-10 fuel design. First, it was noted that the local fuel rod power peaking for SSE-2 Cycle-9 fuel bundles exceeded the range of the ANFB correlation as stated in Reference 4 (local peaking < 1.3). In addition, a flow dependent bias in the ANFB correlation was identified which resulted in the nonconservative overprediction of the measured critical power at low flows. Both of these effects are outside the presently approved SPC SLMCPR and PP&L OLMCPR methodologies.

In order to address these concerns, the methodology used to determine the SLMCPR and the transient ΔCPR has been revised for application to SSE-2 in References 3 and 4. The purpose of this review was to evaluate these methodology changes and insure that adequate margin is included in the SSE-2 Cycle-9 OLMCPR. This review does not include those aspects of the methodology which relate to the generic resolution of the identified ANFB concerns. The methodology changes are summarized in Section 2, and the evaluation of the important technical issues raised during this review is presented in Section 3. The Technical Position is given in Section 4.

2.0 SUMMARY OF THE REVISED OLMCPR METHODOLOGY

The form of the ANFB correlation and the definition of the independent variables described in Reference 5 are not changed for application to the new SPC ATRIUM™-10 fuel design. The dependence of the ANFB correlation on the ATRIUM™-10 bundle design is included by adjusting the values of the additive constants (used to determine the local peaking function) to match the critical power test data. SPC has performed a series of critical power tests and determined the ATRIUM™-10 design-specific additive constants. The ANFB correlation when used with these additive constants reproduces the measured critical power to within the correlation standard deviation.

The initial measurements consisted of a series of twelve critical power tests to determine the critical power characteristics of the ATRIUM™-10 fuel bundle. The tests were performed for an axially symmetric cosine power shape and included a set of limiting local power distributions and a range of pressures, flows and inlet subcoolings. The tests were for a 10×10 rod array and included the ATRIUM™-10 water channel and part-length rods. The ANFB correlation predicted the measured critical power to within a standard deviation which was comparable to previous ANFB applications (Reference 5). The ECPRs were evaluated as a function of power, flow, pressure and inlet subcooling and no clear trends or bias were observed.

In order to insure that the ANFB correlation is applicable to asymmetric axial power shapes, SPC performed additional tests for the ATRIUM™-10 fuel bundle including both upskewed and downskewed power shapes. A selected set of cosine tests were repeated for the upskewed and downskewed power shapes to allow comparison and identify any dependence on power shape. These tests indicated a significant dependence of the critical power data on axial power shape which required a revision of the additive constants (determined based on the cosine tests) to insure a conservative critical power calculation. Using these revised additive constants, the ANFB correlation results in a mean critical power underprediction (i.e., ECPR < 1).

3.0 SUMMARY OF THE TECHNICAL EVALUATION

The SPC Topical Report EMF-97-010 (P) provides the basis for application of the ANFB critical power correlation to the ATRIUM™-10 fuel design for Susquehanna reloads. The report includes the results of the ATRIUM™-10 critical power tests, derivation of the additive constants and the determination of the correlation bias and standard deviation relative to the measurement data. The review of the SPC methodology focused on the accuracy of ANFB in reproducing the critical power test data and its applicability to the Susquehanna ATRIUM™-10 reload fuel. The review included several discussions with SPC during the NRC Vendor Inspection (No. 99900081/97-01) at Siemens Power, and with PP&L and SPC during a meeting on March 26, 1997 at the NRC Headquarters in Bethesda, MD. As a result of these discussions and our review of the methodology several important technical issues were raised which required additional information and clarification from SPC and PP&L. This information was requested in Reference 6 and was provided in the PP&L responses included in References 7-10. This evaluation is based on the material presented in the topical report (Reference 4) and in References 7-10. The evaluation of the major issues raised during this review are summarized in the following.

3.1 Application of the ANFB Correlation to Susquehanna ATRIUM™-10 Reload Fuel

3.1.1 Range of Local Power Peaking

The Reference 5 ANFB correlation is applicable to fuel rod arrays for which the local power peaking factor $F_{local} < 1.3$ (ANFB-1125-P(A); Supplement 1, SER Condition 3.3(1)). The Reference 1 ATRIUM™-10 critical power measurements were intended for a similar range of power peaking and were taken for local peakings up to $F_{local} \sim 1.3$ (Table 6.1, Reference 4). However, during the February 1997 NRC inspection of the SPC reload design activities, it was noted that the Susquehanna-2 Cycle-9 reload core includes several fuel bundles with local peaking factors greater than the ANFB maximum of $F_{local} = 1.3$. In response to this concern, SPC has indicated in Reference 4 that in order to account for the increased ANFB correlation uncertainty that occurs for high local peaking the additive constant uncertainty will be increased for rods having $F_{local} > 1.3$.

3.1.2 Flow-Bias in the ANFB Critical Power Predictions

The ANFB correlation database includes measurements for the cosine, downskew and upskew axial power distributions. During the February 1997 NRC inspection, it was noted that, while the cosine data does not include any clear trend versus power, pressure, inlet subcooling or bundle flow, the ANFB predictions of the upskew data indicate a nonconservative flow-dependent bias. For the upskew tests, ANFB conservatively underpredicts the critical power at high flow rates and nonconservatively overpredicts the critical power at low flow rates. In addition, ANFB tends to generally underpredict the downskew test data.

In response to this concern, in Reference 4 (Figure 6.2) SPC has determined the flow-dependent bias in the ANFB predictions. This flow-dependent bias is based on calculation-to-measurement comparisons for the upskew tests and will be applied in all ANFB critical power calculations. The calculated flow-bias does not take credit for the ANFB conservative underprediction at high flows.

3.1.3 Increased Additive Constant Uncertainty at Low Flows

The ANFB upskew tests were intended to determine the correlation dependence on axial power shape and did not include the full range of bundle flows. Consequently, in the very low flow range where test data was not available, an extrapolation of the high-flow upskew data was performed to determine the critical power. In Reference 4, SPC provides an estimate of the increase in additive constant uncertainty introduced by this extrapolation at low flows. In Reference 8 (Response 4) and Reference 9 (Response 3), SPC has indicated that the SLMCPR analysis is insensitive to this uncertainty. In Reference 9 (Response 3), SPC has conservatively increased this additive constant uncertainty by a factor of ~ 2 in the four lowest flow cases (where the uncertainty has an effect). In three of these cases the SLMCPR was unaffected, and in the remaining case (at 50.0 Mlb/hr) the SLMCPR increased by 0.01. SPC has indicated in Reference 10 that for conservatism the SSE-2 Cycle-9 Technical Specification SLMCPR (at 50.0 Mlb/hr) will include this additional 0.01 increase in the SLMCPR.

3.2 Safety Limit MCPR Calculation

The safety limit MCPR provides the uncertainty allowance required to account for the uncertainties in the ANFB correlation and the POWERPLEX-II core monitoring system. The Susquehanna-2 Cycle-9 SLMCPR was determined using the SPC approved methodology (Reference 11). In the SPC methodology, a cycle-specific full core analysis is performed in which both the fresh ATRIUM™-10 and previous cycle fuel bundles are modeled. The flow is calculated based on the bundle-dependent pressure-drop and used in the ANFB critical power calculation. The bundle-to-bundle difference in pressure-drop and flow resulting from the introduction of the ATRIUM™-10 fuel bundles is accommodated by the full core model.

The initial analysis was performed for a relatively small set of reactor statepoints, however, in Reference 9 (Response 1) SPC has expanded the analysis to include the complete set of standard reactor statepoints. The expanded set of calculations did not result in an increase in the SLMCPR. The SLMCPR analysis was performed over a range of core flows from 108 Mlb/hr down to 30 Mlb/hr. The calculation of the critical power for each fuel bundle included the flow-dependent bias and was determined using the individual bundle flow.

The additive constant uncertainty used in the SLMCPR Monte Carlo calculation was increased to account for the increased uncertainty associated with (1) fuel rods having local peaking greater than 1.3 (discussed in Section 3.1.1), and (2) bundle flows in the lower flow range (discussed in Section 3.1.3). The maximum calculated SLMCPR was used to determine the Susquehanna-2 Cycle-9 operating limit MCPR.

3.3 Determination of ΔCPR for AOOs

The inclusion of the ANFB flow bias increases the AOO transient ΔCPR for events which involve a reduction in the hot bundle flow. Since the calculation of the AOO ΔCPR does not affect the transient dynamics, an ad-hoc correction is used to determine the effect of the ANFB flow bias on the transient MCPR. Based on the calculated transient hot bundle flow reduction and the ANFB flow-dependent bias, a transient-dependent ΔCPR adjustment was determined. The required ΔCPR adjustment to account for the ANFB flow bias has been determined for the limiting AOOs and included in the (Reference 3) Susquehanna-2 Cycle-9 Operating Limit MCPR.

3.4 POWERPLEX-II Core Monitoring System

In the standard POWERPLEX-II core monitoring methodology, the critical power determination is based on the measured reactor statepoint variables together with the ANFB critical power correlation. However, in the Susquehanna-2 Cycle-9 application, the ANFB flow-dependent bias is included in the SLMCPR rather than in the POWERPLEX-II ANFB critical power calculation. While the comparison of the core MCPR to the Operating Limit MCPR insures that the correct thermal margin is maintained, this approach results in an erroneous POWERPLEX-II CPR edit which incorrectly increases the MCPR of the ATRIUM™-10 fuel bundles at reduced flows. Since the ATRIUM™-10 fuel bundles are the MCPR limiting bundles during Cycle-9 this will result in a nonconservative core MCPR edit.

In order to insure that this nonconservative MCPR edit is not misinterpreted it is strongly recommended that PP&L take corrective action, such as eliminating the edit or providing a warning to the staff using this edit.

4.0 TECHNICAL POSITION

The Topical Report EMF-97-010 (P), Revision 1, "Application of ANFB to ATRIUM™-10 for Susquehanna Reloads," and supporting documentation provided in References 7–10 have been reviewed in detail. Based on this review, it is concluded that the proposed methodology and the treatment of additive constant uncertainties, as applied in the determination of the Susquehanna-2 Cycle-9 OLMCPR and SLMCPR, are acceptable subject to the condition stated in Section 3 of this evaluation and summarized in the following.

> In order to insure that the nonconservative POWERPLEX-II MCPR edit is not misinterpreted it is strongly recommended that PP&L take corrective action, such as eliminating the edit or providing a warning to the staff using this edit (Section 3.4).

It is important to recognize that this review does not include those aspects of the methodology which relate to the generic resolution of the identified ANFB concerns. The generic resolution of the ANFB concerns will be reviewed separately.

REFERENCES

1. "Susquehanna Steam Electric Station Proposed Amendment No. 166 to License NPF-22: Unit-2 Technical Specification Changes for ATRIUM-10 Fuel," PLA-4527, Letter, R.G. Byram (PP&L) to U.S. NRC, dated December 18, 1996.

2. "Application of ANFB to ATRIUM-10 for Susquehanna Reloads," EMF-97-010 (P), Revision 0, January 1997.

3. "Susquehanna Steam Electric Station Proposed Amendment No. 166 to License NPF-22: Revised ANFB Methodology and Core Flow Dependent MCPR Safety Limits," PLA-4582, Letter, R.G. Byram (PP&L) to U.S. NRC, dated March 12, 1997.

4. "Application of ANFB to ATRIUM-10 for Susquehanna Reloads," EMF-97-010 (P), Revision 1, March 1997.

5. "ANFB Critical Power Correlation," ANF-1125(P)(A), and Supplement 1 and Supplement 2, Siemens Power Corporation - Nuclear Division, April 19, 1990.

6. "Request for Additional Information (RAI) on Revision to License Amendment Request for Minimum Critical Power Safety Limits for Susquehanna Steam Electric Station, Unit 2 (TAC No. M97499)," Letter, C. Poslusny (NRC) to R.G. Byram (PP&L), dated April 9, 1997.

7. "Susquehanna Steam Electric Station Response to NRC Request for Additional Information on Siemens' Report EMF-97-010, Rev. 1," PLA-4595, Letter, R.G. Byram (PP&L) to U.S. NRC, dated March 27, 1997.

8. "Susquehanna Steam Electric Station Response to NRC Request for Additional Information on PP&L's Proposed Amendment No. 166 to License NPF-22: Unit-2 Technical Specification Changes for ATRIUM-10 Fuel," PLA-4605, Letter, R.G. Byram (PP&L) to U.S. NRC, dated April 9, 1997.

9. "Susquehanna Steam Electric Station Addendum to PP&L's Response to NRC Request for Additional Information on PP&L's Proposed Amendment No. 166 to License NPF-22: Unit-2 Technical Specification Changes for ATRIUM-10 Fuel," PLA-4611, Letter, R.G. Byram (PP&L) to U.S. NRC, dated April 16, 1997.

10. "Susquehanna Steam Electric Station Addendum to PP&L's Response to NRC Request for Additional Information on PP&L's Proposed Amendment No. 166 to License NPF-22: Unit-2 Technical Specification Changes for ATRIUM-10 Fuel," PLA-4613, Letter, R.G. Byram (PP&L) to U.S. NRC, dated April 18, 1997.

11. "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," ANF-524 (P) (A), Revision 2, Siemens Power Corporation - Nuclear Division, April 19, 1989.

UNITED STATES NUCLEAR REGULATORY COMMISSION

PENNSYLVANIA POWER AND LIGHT COMPANY

DOCKET NO. 50-388

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 139 to Facility Operating License No. NPF-22 issued to Pennsylvania Power and Light (the licensee), which revised the Technical Specifications for operation of the Susquehanna Steam Electric Station, Unit 2, located in Luzerne County, PA. The amendment is effective as of the date of issuance.

The amendment modified the Technical Specifications to authorize the use of ATRIUM-10 fuel in the reactor for the ninth refueling cycle for this plant.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on March 18, 1997 (62 FR 12859). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (62 FR 24669).

For further details with respect to the action see (1) the application for amendment dated December 18, 1996 as supplemented on February 26, March 12 and 27, April 3, 9, 16, 18, and 24, 1997, (2) Amendment No. 139 to License No. NPF-22, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document room located at the Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

Dated at Rockville, Maryland, this 7th day of May 1997.

FOR THE NUCLEAR REGULATORY COMMISSION

Chester Poslusny

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