November 2, 1990

Docket No. 50-387

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Mr. Harold W. Keiser
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Keiser:

SUBJECT: CYCLE 6 RELOAD, SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 (TAC NO. 77165)

The Commission has issued the enclosed Amendment No. 102 to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station, Unit 1. This amendment is in response to your letter dated July 2, 1990.

This amendment changes Technical Specifications (TS) in support of the cycle 6 reload.

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/

Mohan C. Thadani, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 102 to License No. NPF-14
2. Safety Evaluation

cc w/enclosures:
See next page

[TAC NO. 77165]

PDI-2/LA	PDI-2/PE	PDI-2/PM	PDI-2/D
MO'Brien	SDembek	MThadani	WButler
10/31/90	10/31/90	10/31/90	11/2/90

DF01
111

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

November 2, 1990

Docket No. 50-387

Mr. Harold W. Keiser
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
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Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station
Units 1 & 2

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

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. 102
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 July 2, 1990 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) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 102 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 2, 1990

PDI-2:LA	PDI-2:PE	PDI-2/PM	OGC	PDI-2/D
MO'Brien	SDembek	MThadani		WButler
10/31/90	10/30/90	10/31/90	11/1/90	11/2/90

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 2, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 102

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. The overleaf pages are provided to maintain document completeness.*

REMOVE	INSERT
xxi	xxi
xxii	xxii*
B 2-2	B 2-2
-	-
3/4 2-1	3/4 2-1*
3/4 2-2	3/4 2-2
3/4 2-9	3/4 2-9
3/4 2-9a	3/4 2-9a
3/4 2-10a	3/4 2-10a*
3/4 2-10b	3/4 2-10b
3/4 4-1b	3/4 4-1b
3/4 4-1c	3/4 4-1c
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
-	-
B 3/4 4-1	B 3/4 4-1
B 3/4 4-1a	B 3/4 4-1a
5-5	5-5*
5-6	5-6

LIST OF FIGURES

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LIST OF TABLES

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. XN-NF-524 (A) Revision 1 describes the methodology used in determining the Safety Limit MCPR.

The XN-3 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 XN-3 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. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 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.

ANF fuel is monitored using the XN-3 critical power correlation. ANF has determined that this correlation provides sufficient conservatism to preclude the need for any penalty due to channel bow. The conservatism has been evaluated by ANF to be greater than the maximum expected ΔCPR (0.02) due to channel bow in C-lattice plants using channels for only one fuel bundle lifetime. Since Susquehanna SES is a C-lattice plant and uses channels for only one fuel bundle lifetime, monitoring of the MCPR limit with the XN-3 critical power correlation is conservative with respect to channel bow and addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel shall not exceed the limits shown in Figures 3.2.1-1 and 3.2.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1 or 3.2.1-2, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

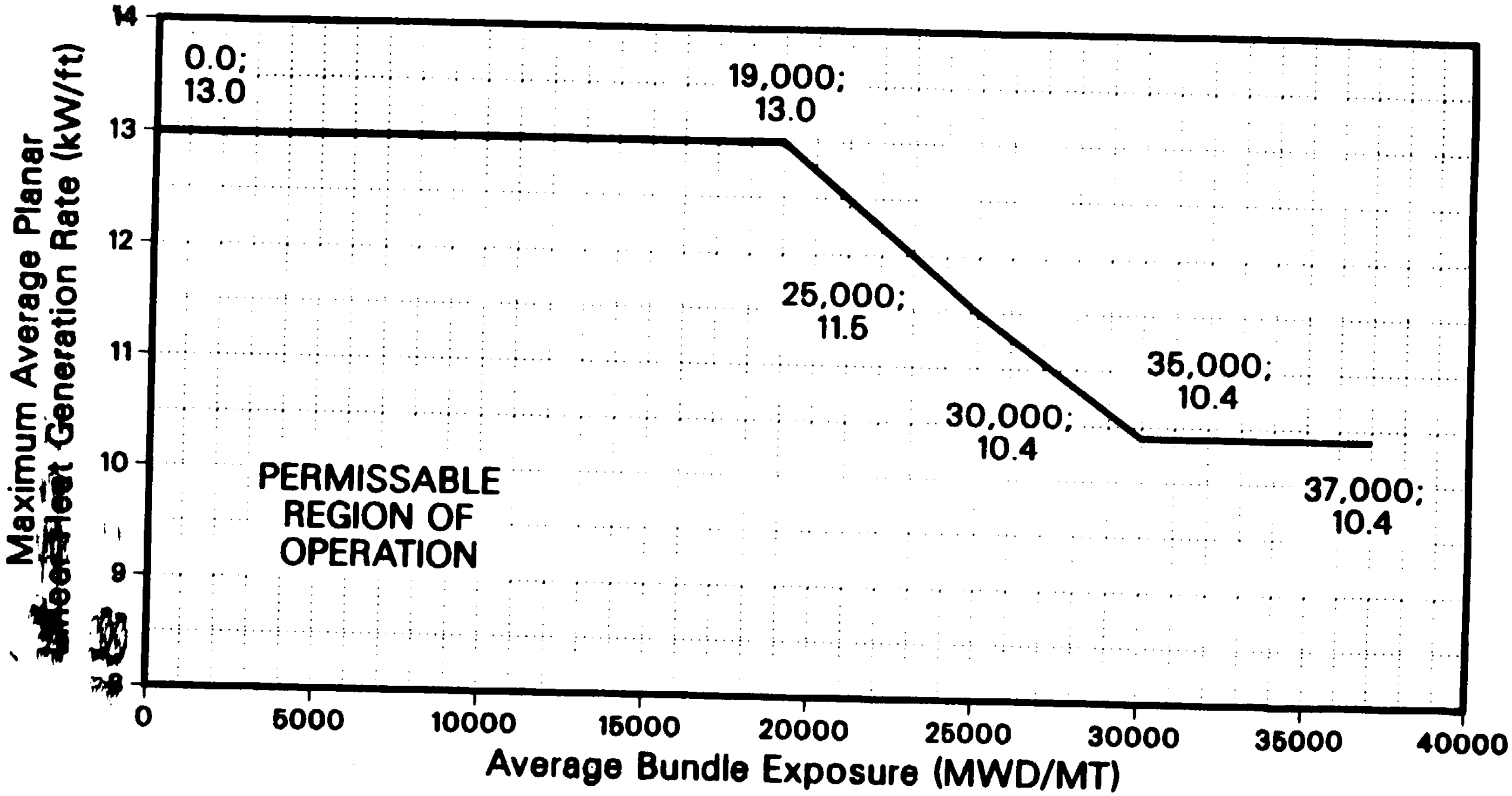
4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, and 3.2.1-2.

a. At least once per 24 hours,

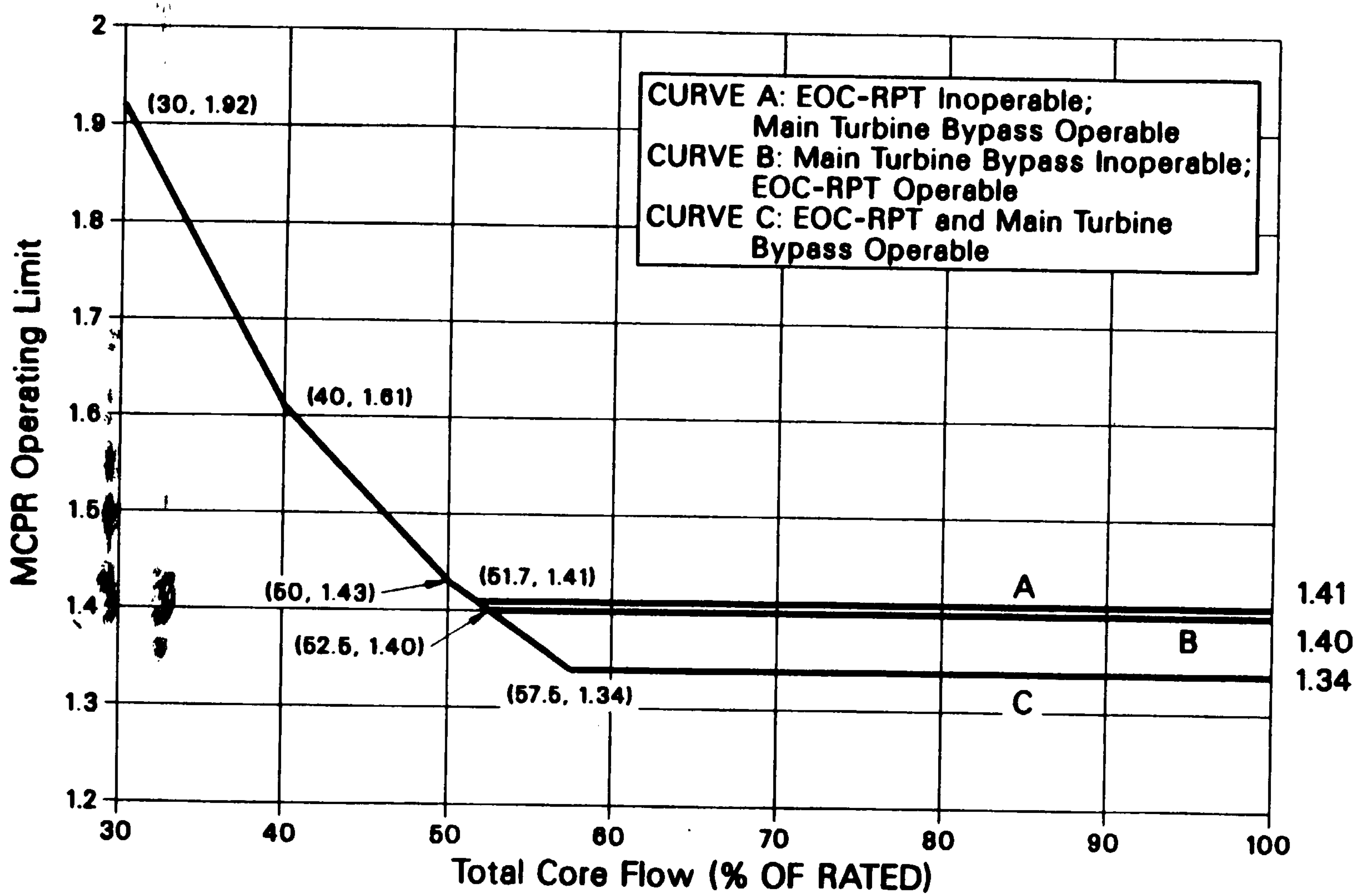
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

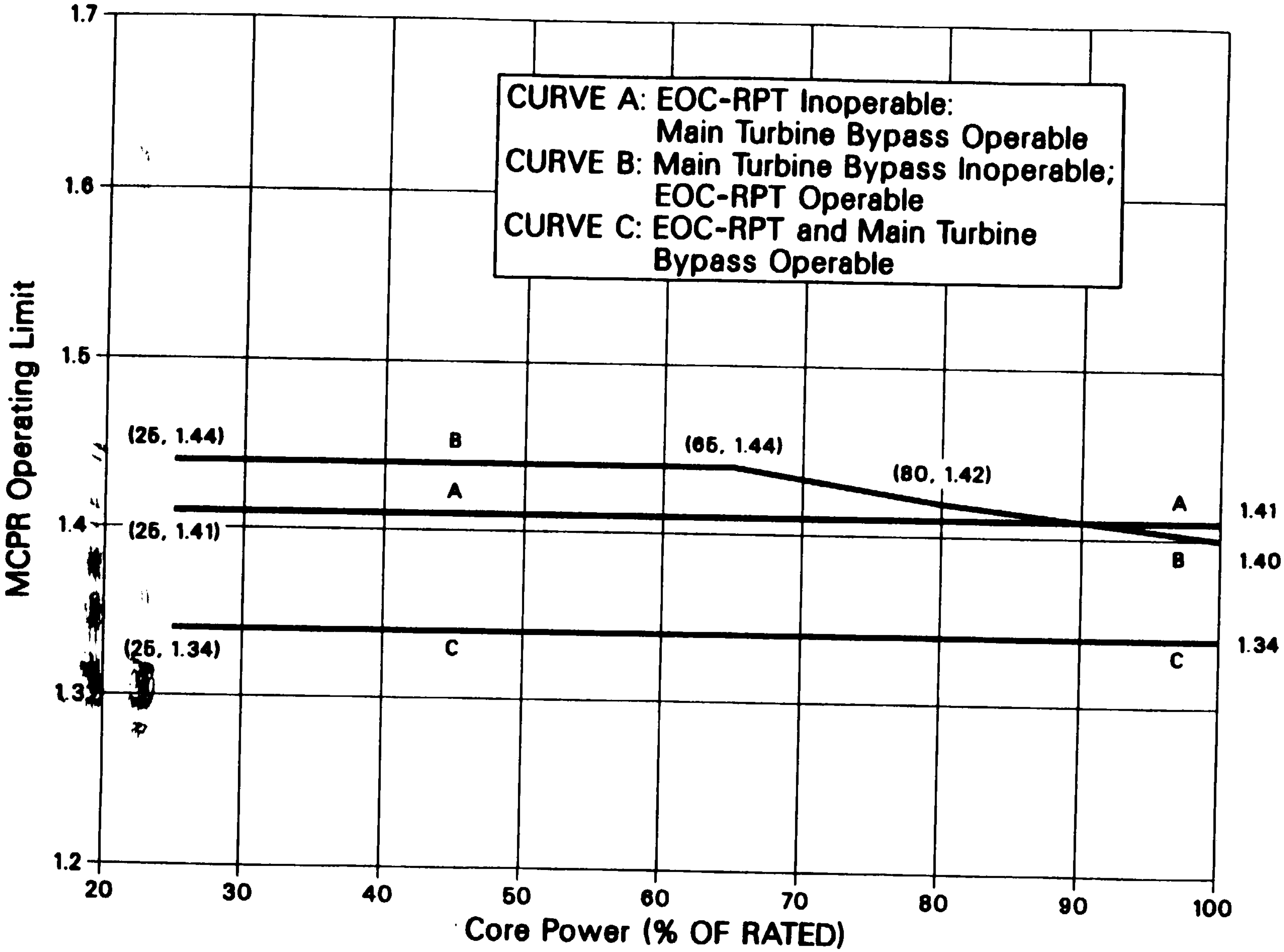
d. The provisions of Specification 4.0.4 are not applicable.



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
ANF 8X8 FUEL
FIGURE 3.2.1-1



FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1



REDUCED POWER MCPR OPERATING LIMIT
Figure 3.2.3-2

3/4.2.4 LINEAR HEAT GENERATION RATE

ANF FUEL

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the LHGR limit determined from Figures 3.2.4-1 and 3.2.4-2.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than orequal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding its applicable limit from Figure 3.2.4-1 or 3.2.4-2, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

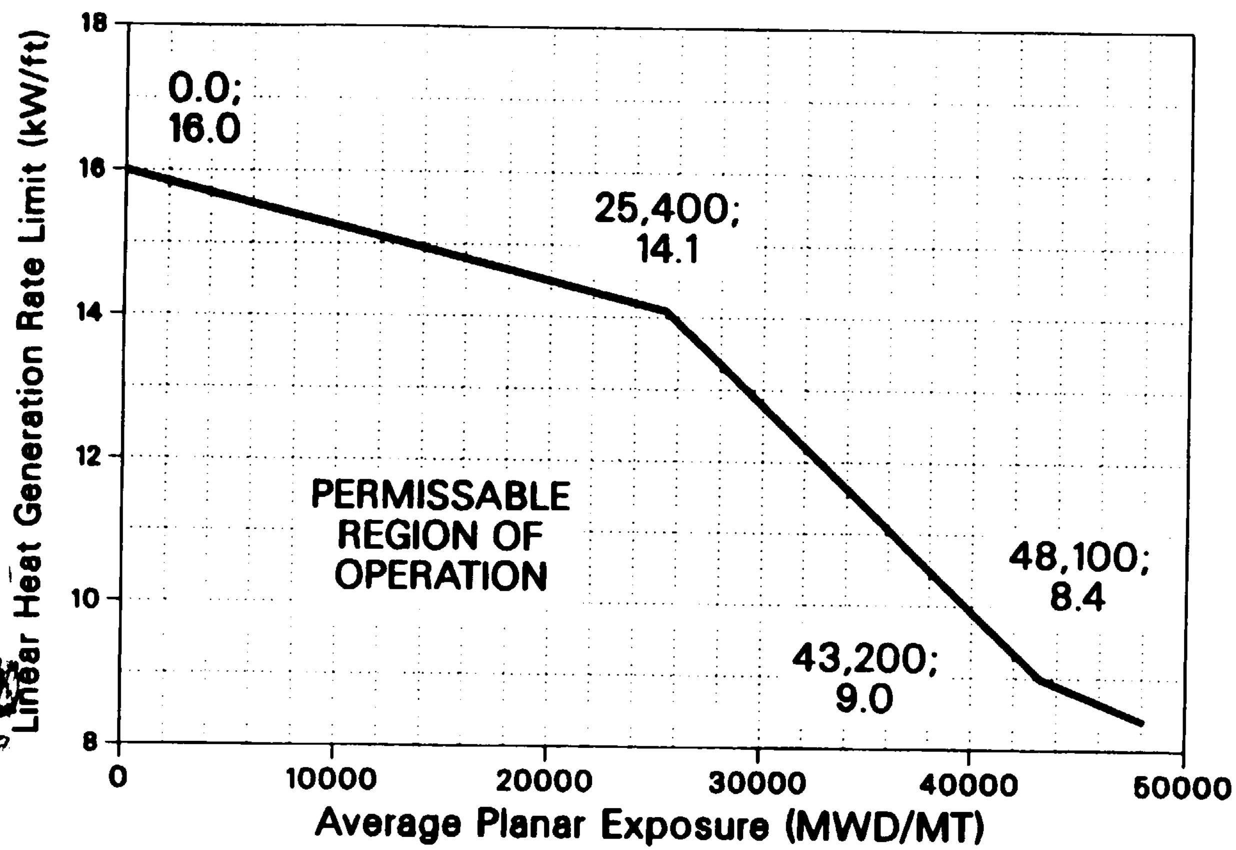
4.2.4 LHGRs shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

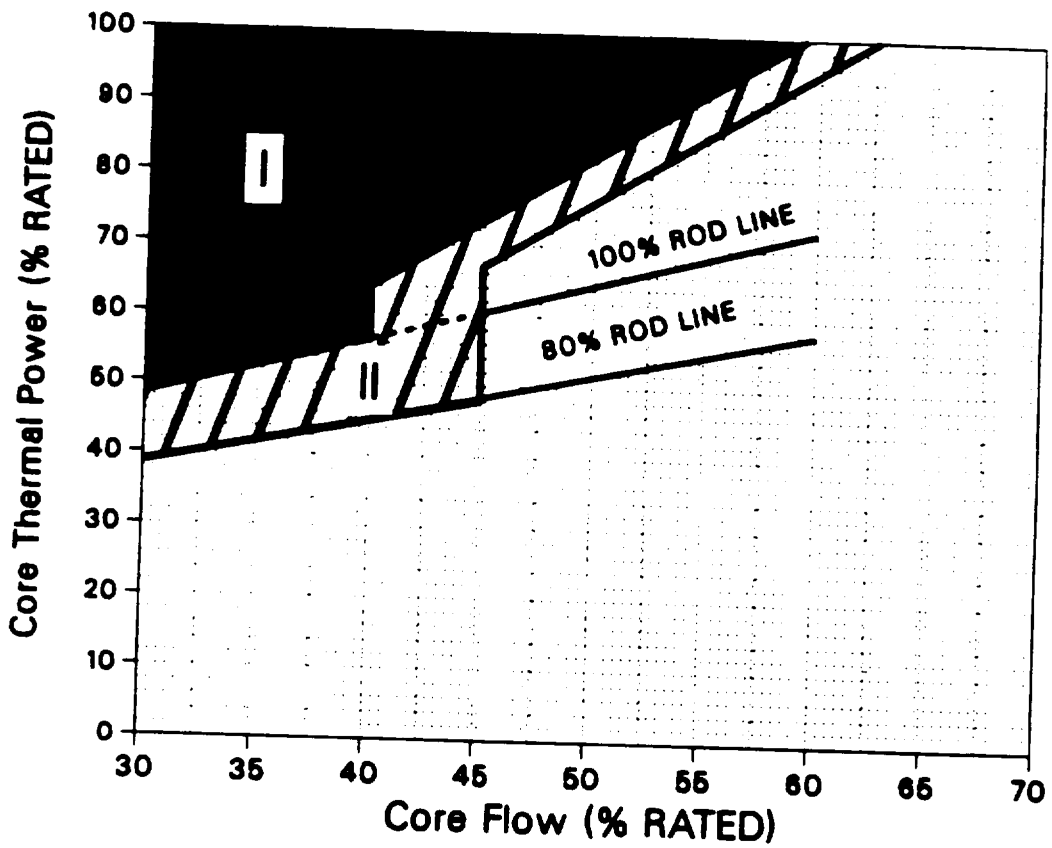
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

d. The provisions of Specification 4.0.4 are not applicable.



LINEAR HEAT GENERATION RATE (LHGR) LIMIT
VERSUS AVERAGE PLANAR EXPOSURE
ANF 8X8 FUEL
FIGURE 3.2.4-1



THERMAL POWER RESTRICTIONS
Figure 3.4.1.1.1

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 1.07.

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
$S \leq (0.58W + 54\%)T$	$S \leq (0.58W + 57\%)T$
$S_{RB} \leq (0.58W + 45\%)T$	$S_{RB} \leq (0.58W + 48\%)T$

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the largest of the following values:

   a. 1.30,

   b. the MCPR determined from Figure 3.2.3-1 plus 0.01, and

   c. the MCPR determined from Figure 3.2.3-2 plus 0.01.

5. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

		Trip Setpoint	Allowable Value
a.	RBM - Upscale	≤ 0.66W + 36%	≤ 0.66W + 39%
b.	APRM-Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

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

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The Technical Specifiation APLHGR for ANF fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is shown in Figures 3.2.1-1 and 3.2.1-2.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, and 3.2.1-2 is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. These models are described in XN-NF-80-19, Volumes 2, 2A, 2B and 2C.

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analyses. In addition, the APRM setpoints must be adjusted to ensure that ≥1% plastic strain and fuel centerline melting do not occur during the worst anticipated operational occurrence (AOO), including transients initiated from partial power operation.

For ANF fuel the T factor used to adjust the APRM setpoints is based on the FLPD calculated by dividing the actual LHGR by the LHGR obtained from Figure 3.2.2-1. The LHGR versus exposure curve in Figure 3.2.2-1 is based on ANF's Protection Against Fuel Failure (PAFF) line shown in Figure 3.4 of XN-NF-85-67(A), Revision 1. Figure 3.2.2-1 corresponds to the ratio of PAFF/1.2 under which cladding and fuel integrity is protected during AOOs.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figures 3.2.3-1 and 3.2.3-2.

The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to an ANF core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and non-pressurization events are described in XN-NF-79-71 and XN-NF-84-105. The principal result of this evaluation is the reduction in MCPR caused by the transient.

Figure 3.2.3-1 defines core flow dependent MCPR operating limits which assure that the Safety Limit MCPR will not be exceeded during a flow increase transient resulting from a motor-generator speed control failure. The flow dependent MCPR is only calculated for the manual flow control mode. Therefore, automatic flow control operation is not permitted. Figure 3.2.3-2 defines the power dependent MCPR operating limit which assures that the Safety Limit MCPR will not be exceeded in the event of a Feedwater Controller Failure, Rod Withdrawal Error, or Load Reject without Main Turbine Bypass Operable initiated from a reduced power condition.

Cycle specific analyses are performed for the most limiting local and core wide transients to determine thermal margin. Additional analyses are performed to determine the MCPR operating limit with either the Main Turbine Bypass inoperable or the EOC-RPT inoperable. Analyses to determine thermal margin with both the EOC-RPT inoperable and Main Turbine Bypass inoperable have not been performed. Therefore, operation in this condition is not permitted.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation

MINIMUM CRITICAL POWER RATIO (Continued)

will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any fuel rod is less than the design linear heat generation even if fuel pellet densification is postulated.

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, for operation in single-loop with an operating MCPR limit ≥1.30, the radiological consequences of a pump seizure accident from single-loop operating conditions are but 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. The boundaries of the regions in Figure 3.4.1.1.1.1-1 are determined using ANF decay ratio calculations and supported by Susquehanna SES stability testing.

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.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

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.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during the loop operation, continued operation is permitted in the single loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

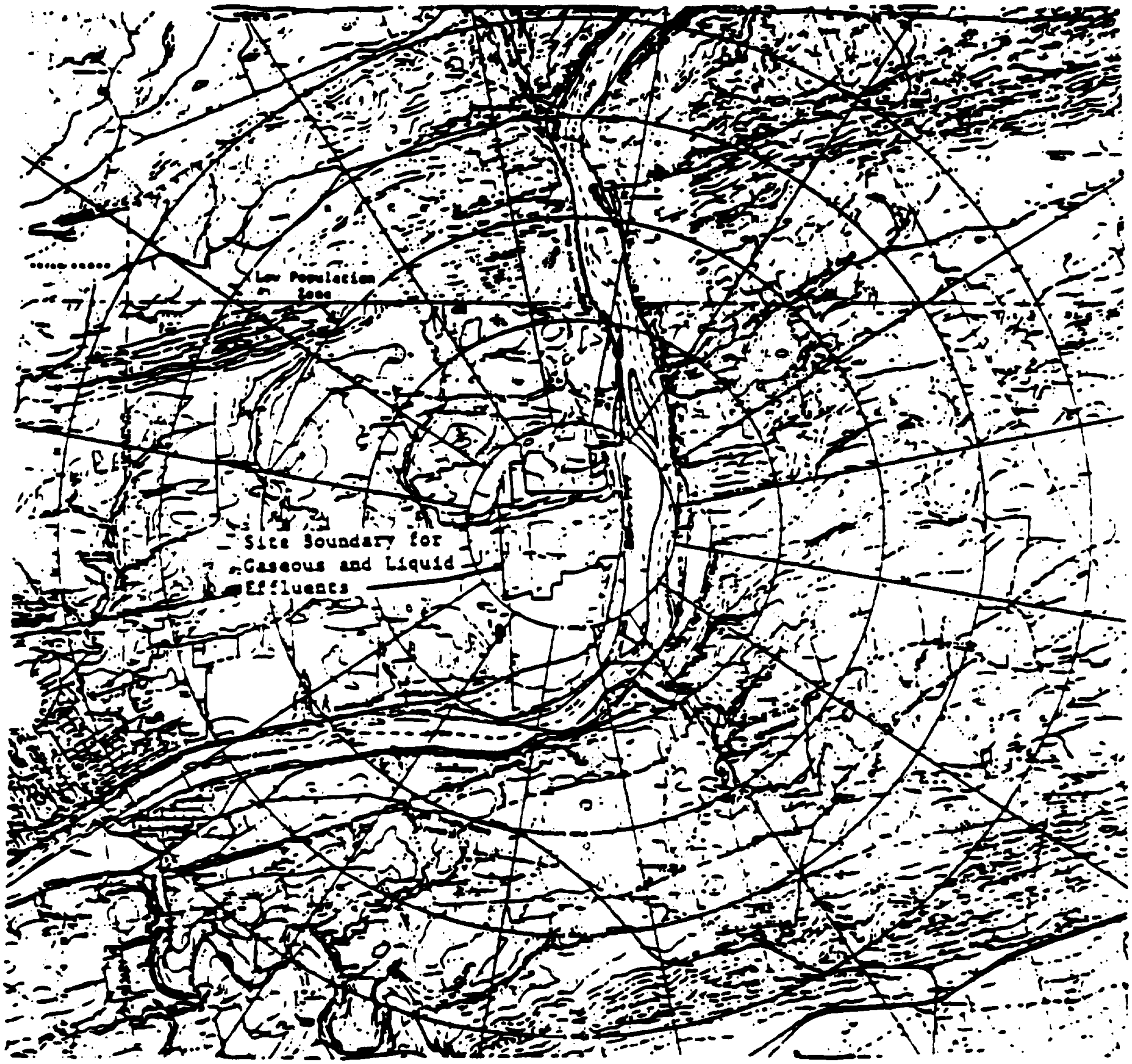


FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED AREAS
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 or 79 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches. Reload fuel shall have a maximum average enrichment of 4.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies consisting of two different designs. The "original equipment" design consists of a cruciform array of stainless steel tubes containing 143 inches of boron carbide (B4C) powder surrounded by a stainless steel sheath. The "replacement" control blade design consists of a cruciform array of stainless steel tubes containing 143 inches of boron carbide (B4C) powder near the center of the cruciform, and 143 inch long solid hafnium rods at the edges of the cruciform, all surrounded by a stainless steel sheath.

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 528°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. NPF-14

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION UNIT NO. 1

1.0 INTRODUCTION

By letter dated July 2, 1990 (Ref. 1), the Pennsylvania Power and Light Company (PP&L) (the licensee) requested an amendment to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station, Unit 1. The proposed amendment would support authorization of Susquehanna Steam Electric Station, Unit No. 1 (Susquehanna 1) operation for Cycle 6 with 9x9 reload fuel supplied by Advanced Nuclear Fuels Corporation (ANF).

The Susquehanna 1 Cycle 6 (S1C6) reload will consist of 220 new ANF-5 9x9 fuel assemblies, 468 irradiated ANF 9x9 assemblies and 76 irradiated ANF 8x8 assemblies. S1C6 will contain no General Electric Company (GE) fuel assemblies. The new 9x9 fuel has similar operating characteristics (mechanical, thermal-hydraulic and nuclear) to the previously used ANF 9x9 reload fuel. In addition to the fuel changes, there will also be a replacement of 50 of the current control rod blades with GE designed Duralife 160C blades. In support of the S1C6 reload, the licensee submitted reports which summarize the reload scope (Ref. 2), the plant transient analyses (Ref. 3), and the design and safety analyses (Ref. 4).

Except for the added discussion of the control rod blade replacement, the analyses, evaluation and results submitted for S1C6 and the reports referenced are similar to those submitted and approved by the NRC staff for the reload for Cycle 5.

2.0 EVALUATION

2.1 Fuel Mechanical Design

The S1C6 core reload will include 220 ANF 9x9 fuel bundles with the designation ANF-5. These reload bundles contain 79 fuel rods and 2 water rods. The 220 fuel bundles will have a bundle average enrichment of 3.52 or 3.21 weight percent uranium-235. The fuel design and safety analysis are described in the Susquehanna 1 specific report PL-NF-90-003 (Ref. 2) and the generic mechanical design report XN-NF-85-67(P)(A), Revision 1 (Ref. 5). The NRC has approved the latter report and issued a Safety Evaluation Report on July 23, 1986 (Ref. 6).

Table 2.1 of XN-NF-85-67(P)(A), Revision 1 (Ref. 5) gives the pertinent design data for ANF 9x9 fuel. Neutronic values specific to the S1C6 reload are given in Table 4.1 of ANF-90-050 (Ref. 4). The burnable poison fuel rods contain 4.0 or 5.0 weight percent gadolinia. The analyses for S1C6 support fuel bundle discharge exposures of 37,000 MWd/MTU for ANF 8x8 fuel and 40,000 MWd/MTU for ANF 9x9 fuel. The discharge exposures for these fuel types are based on the approved ANF topical report XN-NF-82-06(P)(A), Supplement 1, Revision 2 (Ref. 7). Based on our review of the information presented, we find the mechanical design of the ANF 9x9 fuel for the S1C6 reload to be acceptable.

For the S1C6 ANF 9x9 reload fuel, calculation of the fuel rod internal pressure was done in accordance with acceptance criteria cited by ANF in Reference 6. The evaluation was performed with the RODEX2A computer code which has been reviewed and approved by the staff (Ref. 8). The staff has concluded that the acceptance criteria for rod internal pressure can be fully met throughout the entire expected irradiation life of the 9x9 fuel.

A figure of LHGR limit versus planar exposure (MWd/MTU) for the ANF 9x9 fuel is incorporated into the Susquehanna 1 Technical Specifications. This figure was previously approved to reflect the design values which have been reviewed and approved for the ANF 9x9 fuel in connection with the staff's review of XN-NF-85-67(P), Revision 1 (Ref. 5). Based on the results of the generic review, the staff finds the current LHGR limits for the 9x9 fuel to be applicable for the new 9x9 fuel and to be acceptable.

The currently approved exposure limit (35,000 MWd/MTU) for the ANF 8x8 fuel remaining in the core will be exceeded during Cycle 6. ANF has provided (Ref. 9) an analysis justifying the extension of the burnup limit to 37,000 MWd/MTU. This analysis uses approved methodology and acceptance limits and the result is acceptable.

The licensee has discussed the mechanical response of the ANF 9x9 fuel assembly design during LOCA-seismic events in Appendix B of Reference 4. The discussion includes a comparison of the physical and structural properties of the ANF 9x9 fuel and the GE 8x8 fuel. The staff has reviewed this information in connection with a previous review (see Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 31 to Facility Operating License No. NPF-22 dated October 3, 1986). The staff has confirmed that the physical and structural characteristics of the ANF and GE fuel assemblies are sufficiently similar so that the mechanical response to design LOCA-seismic events is essentially the same. Based on the considerations discussed above, we conclude that the original analysis is applicable to Susquehanna 1 and the analysis indicating that the design limits are not exceeded is acceptable.

2.2 Control Rod Blades

PP&L intends to replace up to 50 of the original equipment control rod blades for S1C6 to meet the commitment, in response to Bulletin 79-26, Revision 1, to limit the B^{10} depletion to no more than 34 percent. The replacement will be General Electric (GE) Duralife 160C blades. They are designed to eliminate B_4C tube cracking and increase blade life. They have improved B_4C tube material, hafnium rods at the blade edge, additional B_4C tubes, increased sheath thickness and other mechanical design improvements. They are about 16 pounds heavier than current Susquehanna blades. With the exception of the improved crevice-free structure and an extended handle, these blades are equivalent to the NRC approved Hybrid I Control Blade Assembly (Ref. 10). The mechanical aspects of the crevice-free structure have been approved by the NRC (Ref. 11). GE has analyzed the blade neutronics using the same methodology as was used for the Hybrid I design. The Duralife 160C blade has a slightly larger reactivity worth than original Susquehanna blades, but it is within the criterion of nuclear interchangeability. The blades weigh less than a D lattice blade (Susquehanna is a C lattice) and the basis of the control rod drop accident drop velocity (which assumes a D lattice rod) remains valid. The scram times associated with the blade are not significantly different than for current blades, and there is a considerable margin to TS scram speed limits. The staff review of these blades concludes that they are acceptable for use in S1C6.

2.3 Nuclear Design

The nuclear design methodology used for S1C6 is that presented in the ANF report XN-NF-80-19(A), Volume 1 and Volume 1 Supplements 1 and 2 (Ref. 12), and the PP&L report PL-NF-87-001-A (Ref. 13), which were reviewed and approved by the staff for application to Susquehanna core reloads.

The beginning of cycle shutdown margin is calculated to be 1.07 percent delta-k/k, and the R factor is zero. Thus the cycle minimum shutdown margin is well in excess of the required 0.38 percent delta-k/k. The Standby Liquid Control System also fully meets shutdown requirements.

The existing new fuel storage calculations are based on k-infinity of the fuel assembly. Based on ANF calculations of 9x9 fuel, an average lattice enrichment of less than 3.95 weight percent uranium-235 and a k-infinity of less than or equal to 1.388 will meet the acceptance criterion of k-effective no greater than 0.95 under dry or flooded conditions. Since the zone average enrichment of the new fuel is 3.44 weight percent uranium-235 and the maximum cold, uncontrolled, beginning-of-life k-infinity for the ANF fuel bundle enriched zones is 1.133, the ANF calculations show that the staff's acceptance criterion is met for the new fuel storage vault under dry and flooded conditions. To preclude criticality at optimum moderation conditions, watertight covers and appropriate procedures are used. These are acceptable.

ANF also performed analyses for 9x9 fuel stored in the spent fuel pool. A maximum enriched zone of less than 3.95 weight percent uranium-235 meets the staff acceptance criterion of k-effective no greater than 0.95. Since the ANF-5 9x9 fuel has a zone average enrichment of 3.64 or 3.31 weight percent uranium-235 the staff's acceptance criterion for spent fuel storage is met for the ANF-5 9x9 fuel.

Susquehanna will continue to use the ANF POWERPLEX core monitoring system to monitor core parameters. The system has been in use for a number of cycles for both Susquehanna Unit 1 and Unit 2 and has provided acceptable monitoring and predictive results.

2.4 Thermal-Hydraulic Design

The minimum critical power ratio (MCPR) safety limit for the S1C6 reload was determined by the licensee to be 1.06 for all fuel types. The methodology for S1C6 is based on the ANF methodology in XN-NF-80-19(P)(A), Volume 4, Revision 1 (Ref. 14), which has been approved by the staff. The XN-3 correlation used to develop the MCPR safety limit has been approved for the ANF 9x9 fuel (Ref. 15). ANF has determined that this correlation provides sufficient conservatism such that there is no need for any penalty due to channel bow for S1C6. Susquehanna is a C lattice core and uses channels for only one bundle lifetime. For such cores ANF has determined that the conservatism is greater than the maximum expected delta CPR (critical power ratio). The staff has reviewed the ANF channel bow analyses methodology and it is acceptable for this analyses for S1C6.

The core bypass flow fraction has been calculated to be 10.0 percent of total core flow using the approved methodology described in XN-NF-524(P)(A), Revision 1 (Ref. 16). This is used in the MCPR safety limit calculations and as input to the S1C6 transient analyses and is acceptable.

In response to Bulletin 88-07, Supplement 1 (Ref. 17) on BWR thermal-hydraulic stability, PP&L developed restricted operating regions on the power/flow operating map which were in compliance with the NRC recommendations. Technical Specifications (TS) implementing these regions have been approved by the staff for Susquehanna 1. Stability tests have been conducted in Susquehanna 2 with various amounts of ANF 9x9 fuel from succeeding reloads, including all 9x9 fuel. These have indicated no significant deterioration of decay ratio. Decay ratios were low in all tests. Calculations similar to those setting up the restrictive boundaries were done for S1C6. This resulted in slight modifications of the regions for this cycle. TS implementing the changes have been submitted. This review concludes that the analyses are suitable and the changes to the TS are acceptable.

2.5 Transient and Accident Analyses

Various operational transients could reduce MCPR below the safety limit. The most limiting transients have been analyzed to determine which event could potentially result in the largest reduction in the initial Critical Power Ratio (CPR), that is, the delta CPR. The core wide transient which resulted in the largest delta CPR from a 104 percent power and a 100 percent flow condition is the generator load rejection without bypass event (LRWOB). The delta CPR for this event is 0.28 for ANF 9x9 fuel, which is the most limiting fuel type. When combined with a safety limit MCPR of 1.06 this results in a MCPR operating limit of 1.34 for S1C6. The most limiting local transient, the control rod withdrawal error (CRWE), was analyzed to support a rod block monitor (RBM) setpoint of 108 percent and resulted in a delta CPR of 0.26. The LRWOB and the CRWE events were the most limiting events for S1C6 at rated power and flow conditions. At less than rated power, the feedwater controller failure (FWCF) event is limiting and a curve of MCPR versus power, which is based on the FWCF results, is included in the Technical Specifications as a power dependent MCPR operating limit.

At reduced flow conditions, the recirculation flow controller failure transient (RFIT) is limiting and MCPR operating limits for manual flow control reduced flow operation for S1C6 based on the analysis of this event are provided as a Technical Specification figure of MCPR versus core flow. The calculations of the thermal margin were performed with approved methodology (Ref. 18) and the resulting Technical Specification limiting curves are acceptable.

It was assumed for the above analyses that the turbine bypass system and the end-of-cycle recirculation pump trip (RPT) were operable. Analyses were also performed to determine MCPR operating limits with either of these systems inoperable. This resulted in increased MCPR limits which are also proposed for S1C6. These calculations follow standard procedures and operation within the proposed MCPR operating limits with either the main turbine bypass system inoperable or the end-of-cycle RPT inoperable is acceptable for S1C6. Compliance with overpressurization criteria was demonstrated by analysis of the main steam isolation valve (MSIV) closure event assuming MSIV position switch scram failure, an MSIV closure time of 2.0 seconds and six safety-relief valves out-of-service. Maximum vessel pressure was 1,312 psig, within the limit of 1,375 psig. The calculation was done with approved methodology and the results are acceptable.

The LOCA analyses for the Susquehanna plants (Ref. 19) was performed for a full core of ANF 9x9 fuel and is applicable for the S1C6 residual and reload ANF fuel. In addition, to support the increased burnup for the ANF 8x8 fuel, ANF performed an additional LOCA heatup calculation at 40,000 MWd/MTU. These analyses have covered an acceptable range of conditions, have been performed with approved methodology and the resulting Technical Specification MAPLHGR values for the ANF fuel remain acceptable.

The control rod drop accident (CRDA) was analyzed with approved ANF methodology (Ref. 12). The maximum fuel rod enthalpy was 205 cal/gm, which is well below the design limit of 280 cal/gm, and less than 600 fuel rods exceed 170 cal/gm, which is less than the 770 rods assumed in the Susquehanna FSAR analysis. To ensure compliance with the CRDA analysis assumptions, control rod sequencing below 20 percent core thermal power must comply with GE's banked position withdrawal sequencing constraints (Ref. 20). The staff concludes that the analysis and results for the S1C6 CRDA are acceptable.

2.6 Single Loop Operation (SLO)

Current Technical Specifications for Susquehanna Unit 1 permit plant operation with a single recirculation loop out-of-service for an extended period of time. Analyses for S1C6 (Ref. 4) show that the MCPR Safety Limit must be increased by 0.01 because of the increased measurement uncertainties. The pump seizure event is more severe under SLO than under two-loop operation, assuming pump seizure of the operating loop. This is the limiting event over most of the power and flow operating region for SLO. ANF analyzed the pump seizure event on a generic basis for the Susquehanna Units. Calculations were done for several cycles of operation for the Susquehanna Units. The calculated delta CPRs were used to determine a conservative bounding delta CPR. This, combined with a minimum CPR value which would conservatively meet pump seizure accident radiological guidelines of a small fraction of 10 CFR 100 guidelines resulted in a MCPR operating limit of 1.30 for SLO. This is incorporated as a limit in the SLO TS. This analysis used approved methods and the result is acceptable.

Previous analyses reported by the licensee (Refs. 21 and 22) have shown that other events which could be affected by SLO were non-limiting when analyzed under SLO conditions. SLO for S1C6 must maintain the 80 percent recirculation pump speed restriction because of the previous GE vessel internal vibration analysis, as discussed in Reference 21. This requirement is already present in the Technical Specifications and is unchanged by this amendment.

3.0 TECHNICAL SPECIFICATION CHANGES

The following Technical Specification (TS) changes have been proposed for operation of S1C6.

(1) TS 3/4.2.1 -- Figure 3.2.1-1 is changed to reflect the approved burnup extension of ANF 8x8 fuel to 37,000 MWd/MTU for average bundle exposure, which was previously discussed. This is acceptable.

(2) TS 3/4.2.3 -- Figures 3.2.3-1 and -2 are changed to reflect the new calculations of flow and power dependent MCPR operating limits using the parameters of S1C6. As previously discussed, these analyses have been approved and the changes are acceptable.

(3) TS 3/4.2.4 -- Figure 3.2.4-1 is changed to reflect the approved burnup extension of ANF 8x8 fuel to 42,000 MWd/MTU for average planar exposure, which was previously discussed. (The increase from 37,000 to 42,000 for average bundle versus average planar reflects the axial exposure peaking factor.) This change is acceptable.

(4) TS 3/4.4.1 -- Figure 3.4.1.1.1-1 is changed to reflect the calculated changes in the regional stability boundaries, as was previously discussed. The change is acceptable.

(5) TS 3/4.4.1 -- The MCPR operating limit for SLO is changed to 1.30. As was previously discussed, the analysis and results for this change is acceptable.

(6) TS 5.3.1 -- This change removes references to fuel assembly types from the initial core loading which are no longer present. It is acceptable.

(7) TS 5.3.2 -- This change recognizes the presence of the replacement control rod blades. It is acceptable.

In addition there are several administrative and descriptive changes to the Bases reflecting removal of errors or the reasons for the TS changes discussed above. These include Bases 2.12, 3/4.2.1 and .3 and 3/4.4.1. These changes are acceptable.

4.0 TECHNICAL CONCLUSIONS

The staff has reviewed the reports submitted for the Cycle 6 operation of Susquehanna Unit 1 and concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 33992) on August 22, 1990 and consulted with the Commonwealth of Pennsylvania. No public comments were received, and the Commonwealth of Pennsylvania did not have any comments.

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

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REFERENCES

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

2. PL-NF-90-003, "Susquehanna SES Unit 1-Cycle 6: Reload Summary," June 1990.

3. ANF-90-049, "Susquehanna Unit Cycle 6: Plant Transient Analysis," May 1990.

4. ANF-90-050, "Susquehanna Unit 1 Cycle 6 Reload Analysis: Design and Safety Analyses," May 1990.

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. Letter from G. C. Lainas (NRC) to G. N. Ward (ENC), "Acceptance for Referencing of Licensing Topical Report XN-NF-85-67(P), Revision 1, 'Generic Mechanical Design Report for Exxon Nuclear Jet Pump BWR Reload Fuel,'" July 23, 1986.

7. 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.

8. Letter from G. c. Lainas (NRC) to G. N. Ward (ANF), "Acceptance for Referencing of Licensing Topical Report XN-NF-85-74(P)," June 24, 1986.

9. ANF-90-018(P), Revision 1, "Susquehanna Unit 1 8x8 Extended Burnup Design Report,: June 1990.

10. NEDE-22290-A, Supplement 1, "Safety Evaluation of the General Electric Hybrid I Control Rod Assembly for the BWR 4/5 C Lattice," July 1985.

11. NEDE-22290-A, Supplement 3, "Safety Evaluation of the General Electric Duralife 230 Control Rod Assembly," May 1988.

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

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

14. 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.

15. Letter (RAC:058:88) from R. A. Copeland (ANF) to M. W. Hodges (NRC), "Void History Correlation," September 13, 1988.

16. Letter from G. C. Lainas (NRC) to G. N. Ward (ANF), "Acceptance for Referencing of Licensing Topical Report XN-NF-85-74(P), 'RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation,'" June 24, 1986.

17. NRCB-88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors," USNRC Bulletin, December 30, 1988.

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

19. XN-NF-86-65, "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for 9x9 Fuel," May 1986.

20. NEDO-21231, "Banked Position Withdrawal Sequence," General Electric Company, January 1977.

21. Letter (PLA-2885) from PP&L to NRC, "Proposed Amendment 52 to License No. NPF-22," June 30, 1987.

22. Letter (PLA-2935) from PP&L to NRC, "Additional Information on Proposed Amendment 52 to License No. NPF-22," October 30, 1987.