May 15, 1989

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: TECHNICAL SPECIFICATION CHANGES SUPPORTING CYCLE 5 OPERATIONS (TAC NO. 72094)

RE: SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 90 to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station, Unit 1. This amendment is in response to your letter dated February 2, 1989.

This amendment changes the Technical Specifications to support Cycle 5 operations with Advanced Nuclear Fuel Corporation's 9X9 reload fuel.

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

Enclosures:
1. Amendment No. 90 to License No. NPF-14
2. Safety Evaluation

cc w/enclosures:
See next page

[SUSQ RELOAD]

PDI-2/LA	PDI-2/PM	PDI-2/D
MO'Brien	MThadani:mr	WButler
5/11/89	5/12/89	5/15/89

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PDR ADOCK 05000387
P PDC

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

May 15, 1989

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Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
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See next page

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. 90
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 February 2, 1989 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. 90 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.

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P PDC

3. This license amendment is effective as of its date of issuance to be implemented upon startup for Cycle 5 operations currently scheduled for June 2, 1989.

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: May 15, 1989

PDI-2/LA
MBrown
5/15/89

PDI-2/PM
MThadani:mr
5/2/89

OGC
5/9/89

PDI-2/D
WButler
5/15/89

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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: May 15, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 90

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
iii	iii*
iv	iv
xxi	xxi
xxii	xxii*
2-3	2-3*
2-4	2-4
B 2-1	B 2-1
B 2-1a	B 2-1a*
B 2-2	B 2-2
-	-
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-5a	3/4 2-5a*
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-9	3/4 2-9
3/4 2-9a	3/4 2-9a
3/4 2-10	3/4 2-10
-	-
3/4 2-10a	3/4 2-10a
3/4 2-10b	3/4 2-10b
3/4 2-10c	3/4 2-10c
-	-

ATTACHMENT TO LICENSE AMENDMENT NO. 90

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

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3/4 4-1b	3/4 4-1b*
3/4 4-1c	3/4 4-1c
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2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
(1) Flow Biased	≤ 0.58 W+59%,# with a maximum of	≤ 0.58 W+62%,# wit a maximum of
(2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches above instrument zero*	≥ 11.5 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 10% closed	≤ 11% closed
6. Main Steam Line Radiation - High	≤ 7.0 x full power background	≤ 8.4 x full power background
7. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 88 gallons	≤ 88 gallons
b. Float Switch	≤ 88 gallons	≤ 88 gallons
9. Turbine Stop Valve - Closure	≤ 5.5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 460 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

*See Bases Figure B 3/4 3-1.

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

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated 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 Specifications 2.1.2 for ANF 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. XN-NF-524(A) Revision 1).

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the XN-3 correlation is valid for critical power calculations at pressures greater than 580 psig and bundle mass fluxes greater than 0.25×10^6 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 ANF 9x9 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For the ANF 8x8 fuel, the minimum bundle flow is greater than 28,000 lbs/hr. For all 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 critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lbs/hr-ft^2 is 3.35 Mwt or greater. At

2.1.1 THERMAL POWER, Low Pressure or Low Flow (Continued)

25% thermal power a bundle power of 3.35 Mwt corresponds to a bundle radial peaking factor of greater than 3.0 which is significantly higher than the expected peaking factor.

Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressures below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

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

The Safety Limit MCPR assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. 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.

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 as a function of AVERAGE BUNDLE EXPOSURE for ANF 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.

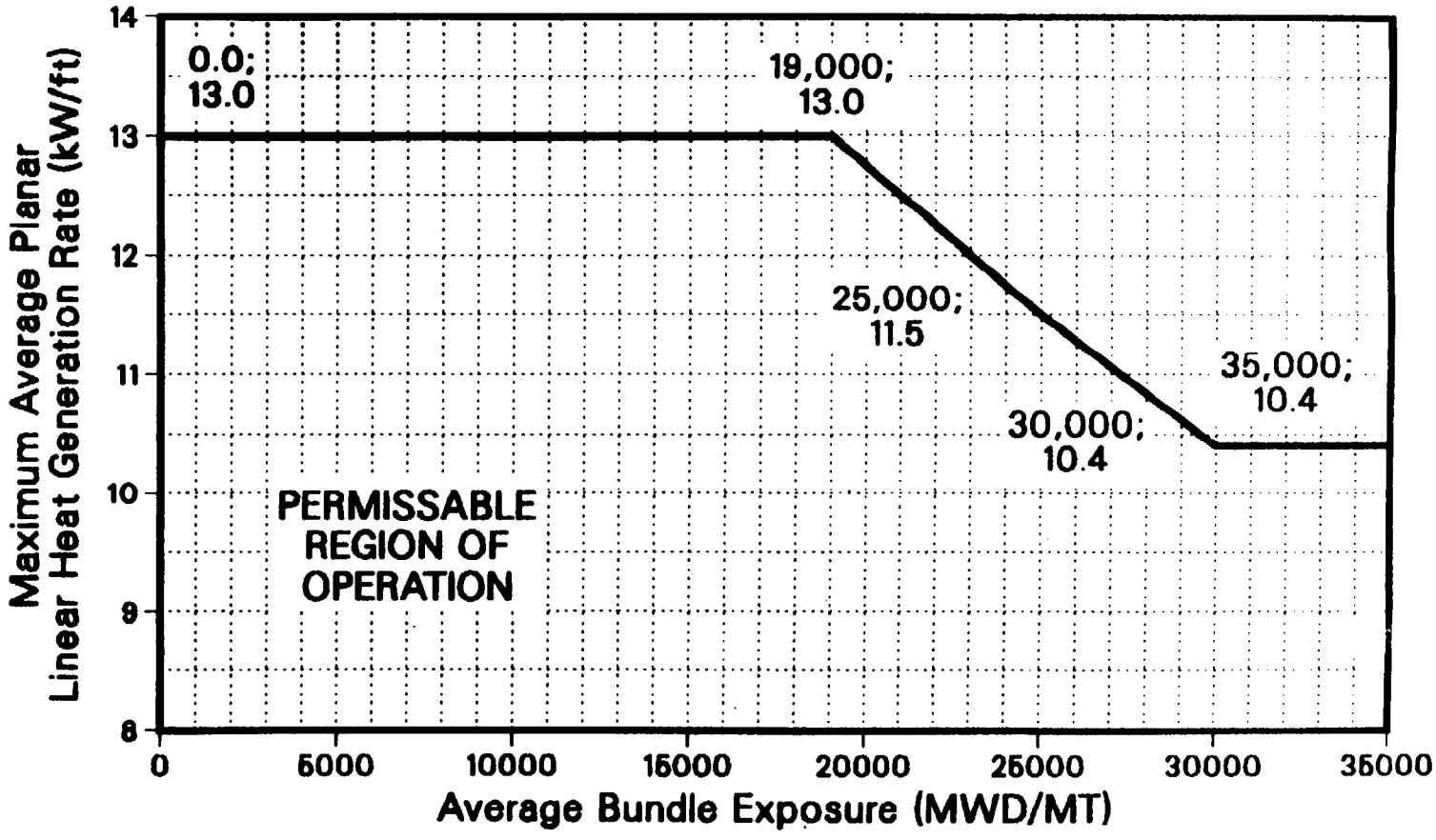
a. At least once per 24 hours,

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

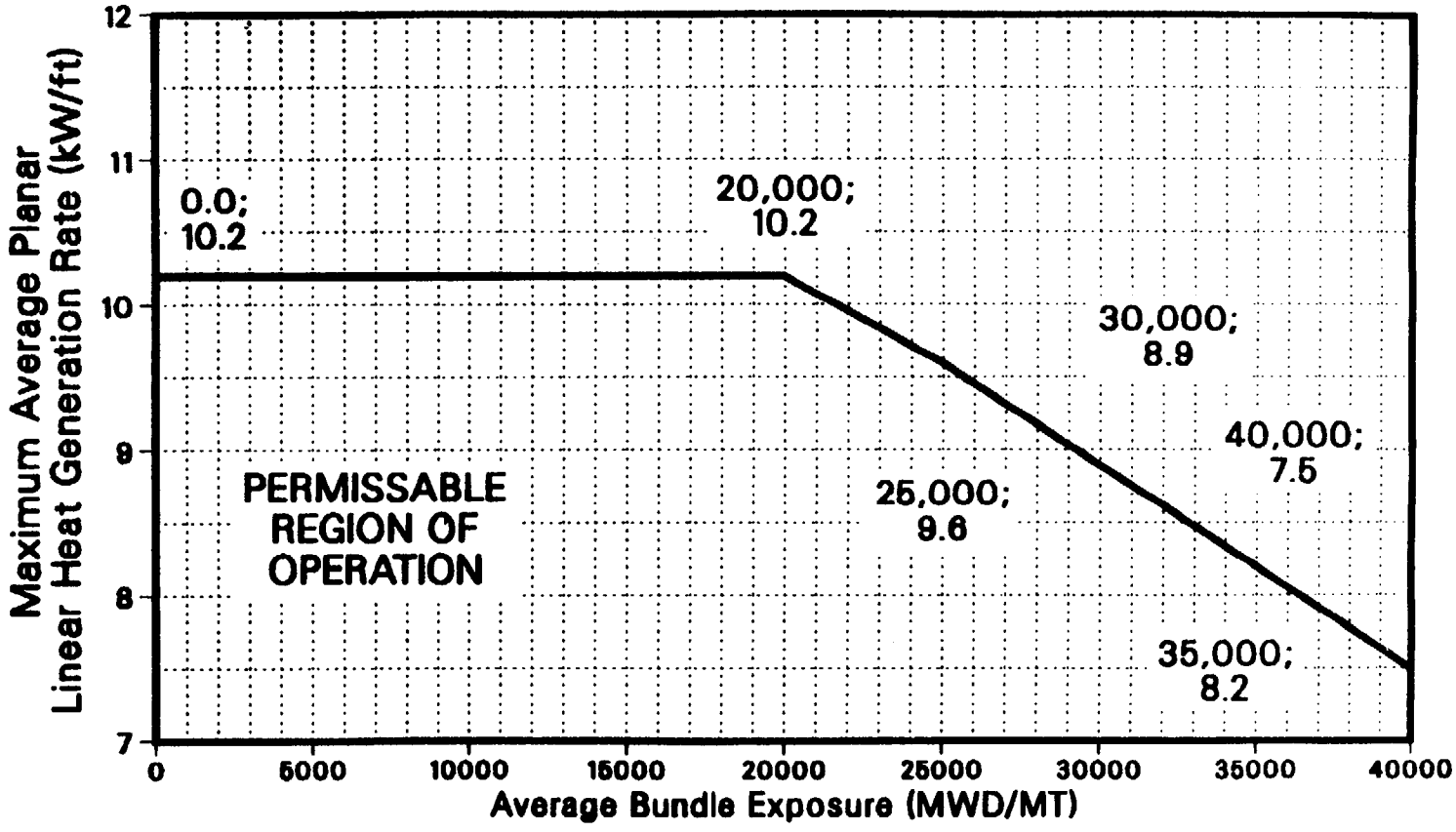
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

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



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



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
ANF 9X9 FUEL
FIGURE 3.2.1-2

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3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

Trip Setpoint#	Allowable Value#
$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. The FLPD for ANF fuel is the actual LHGR divided by the LINEAR HEAT GENERATION RATE from Figure 3.2.2-1.

T is always less than or equal to 1.0.

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

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

#See Specification 3.4.1.1.2.a for single loop operation requirements.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the MFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

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 MFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

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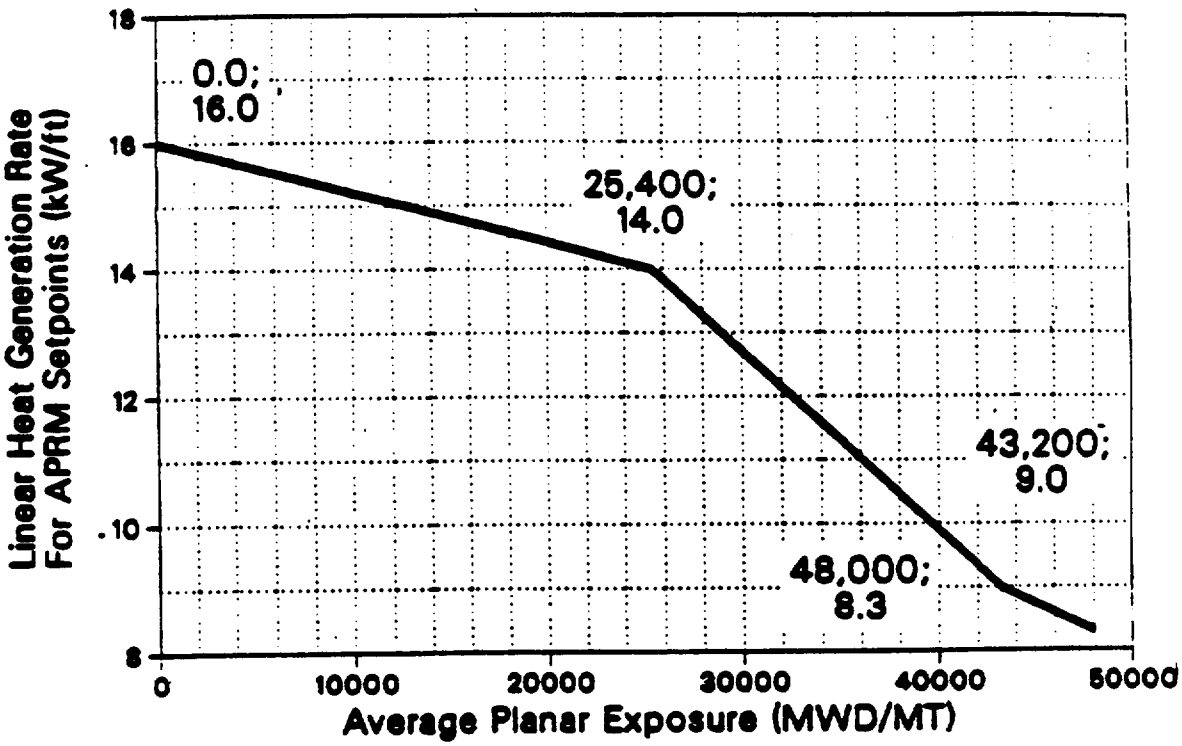


LINEAR HEAT GENERATION RATE FOR APRM SETPOINTS
VERSUS AVERAGE PLANAR EXPOSURE
ANF FUEL
FIGURE 3.2.2-1

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the greater of the two values determined from Figure 3.2.3-1 and Figure 3.2.3-2

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

ACTION:

With MCPR less than the applicable MCPR limit determined above, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

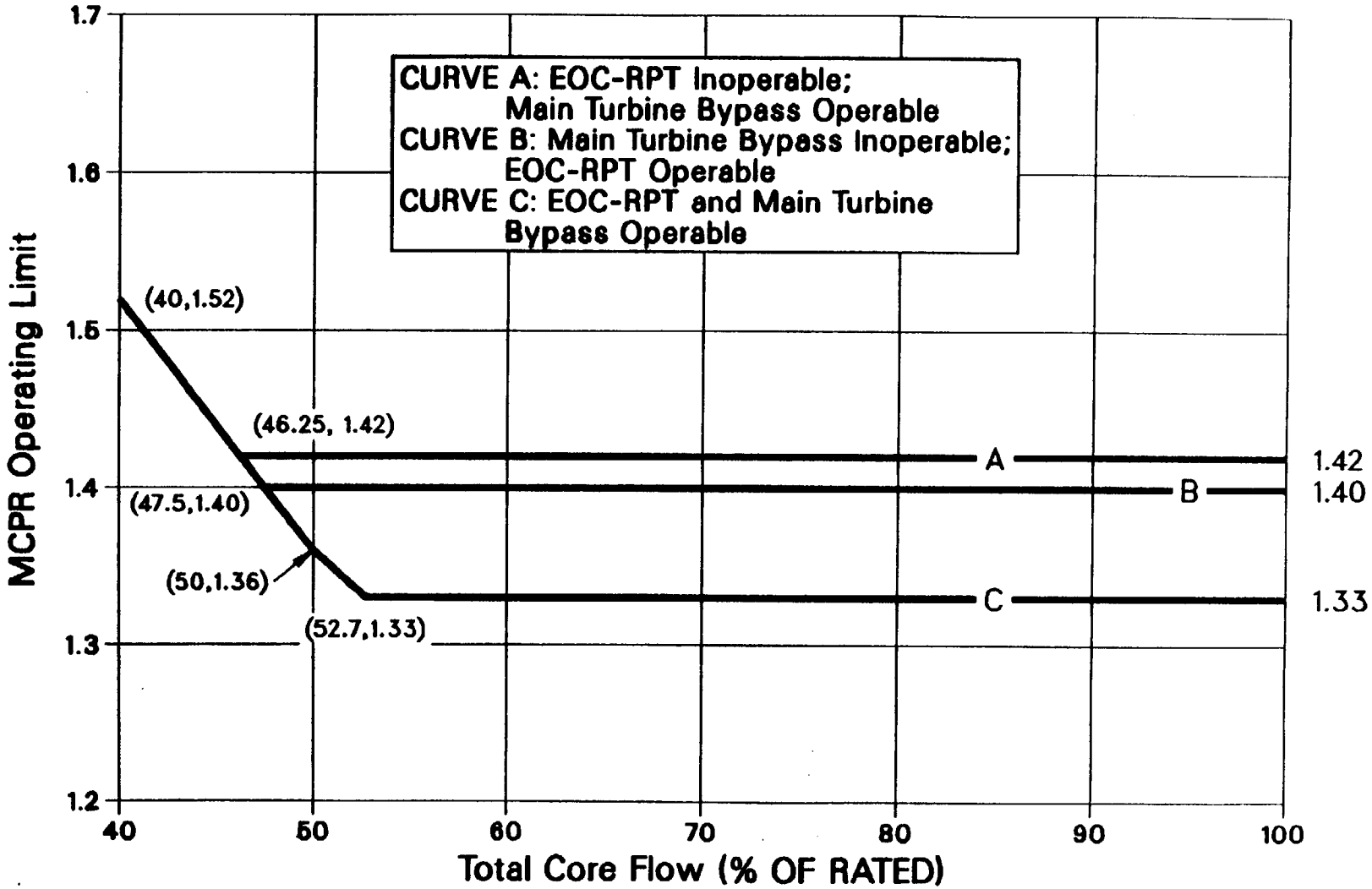
4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit determined from Figure 3.2.3-1 and Figure 3.2.3-2:

a. At least once per 24 hours,

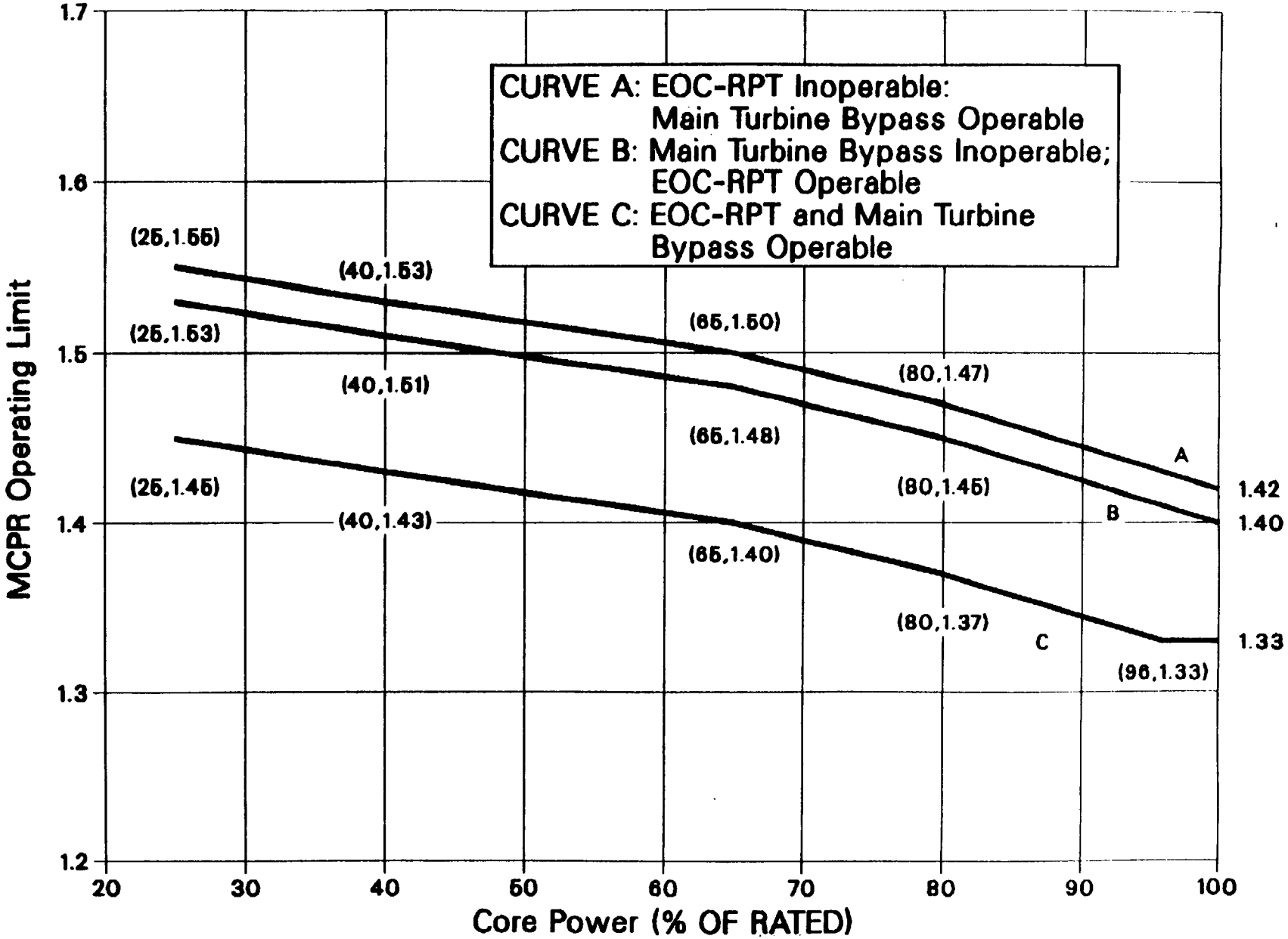
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

d. The provisions of Specification 4.0.4 are not applicable.



FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1



REDUCED POWER MCPR OPERATING LIMIT
Figure 3.2.3-2

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3/4.2.4 LINEAR HEAT GENERATION RATE

ANF FUEL

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) for ANF fuel 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

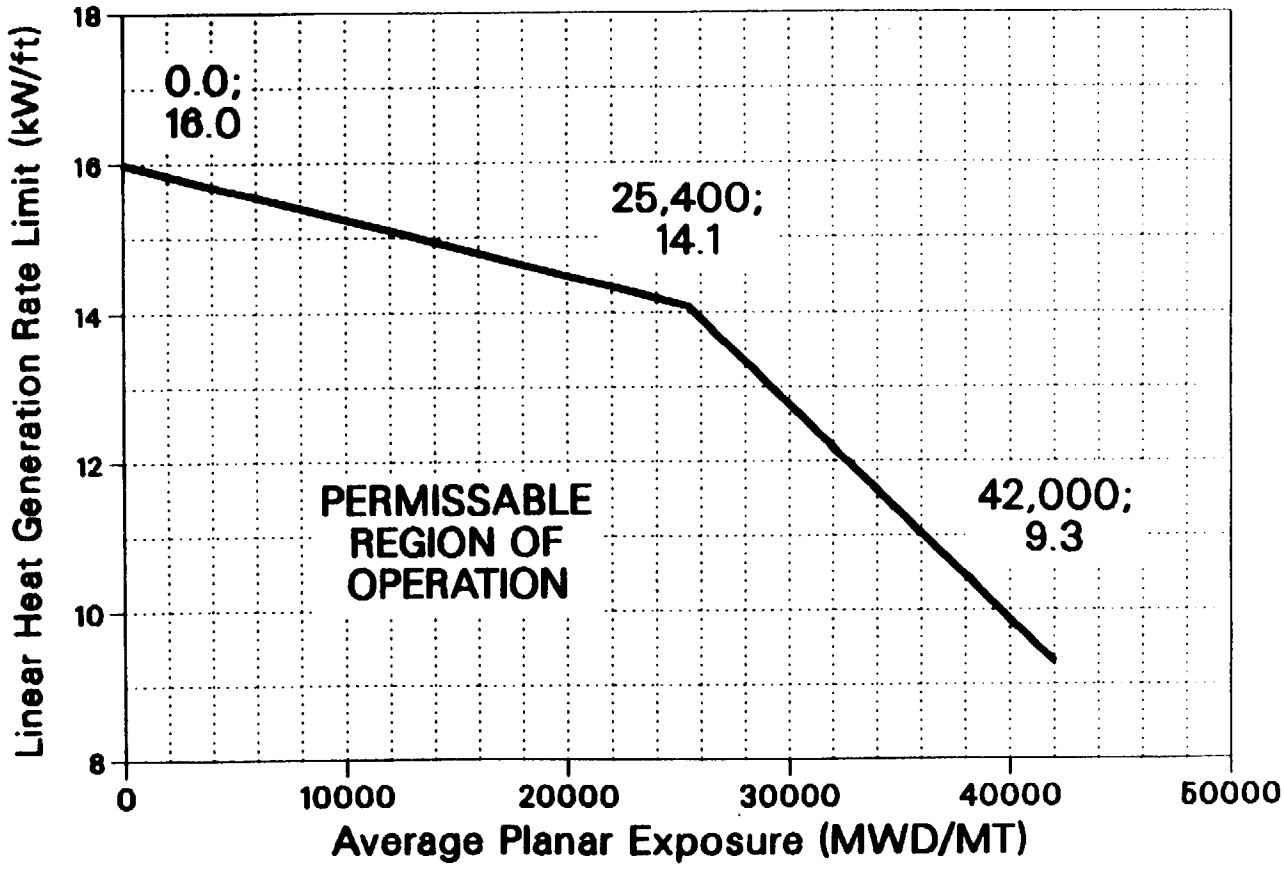
4.2.4 LHGRs for ANF fuel shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

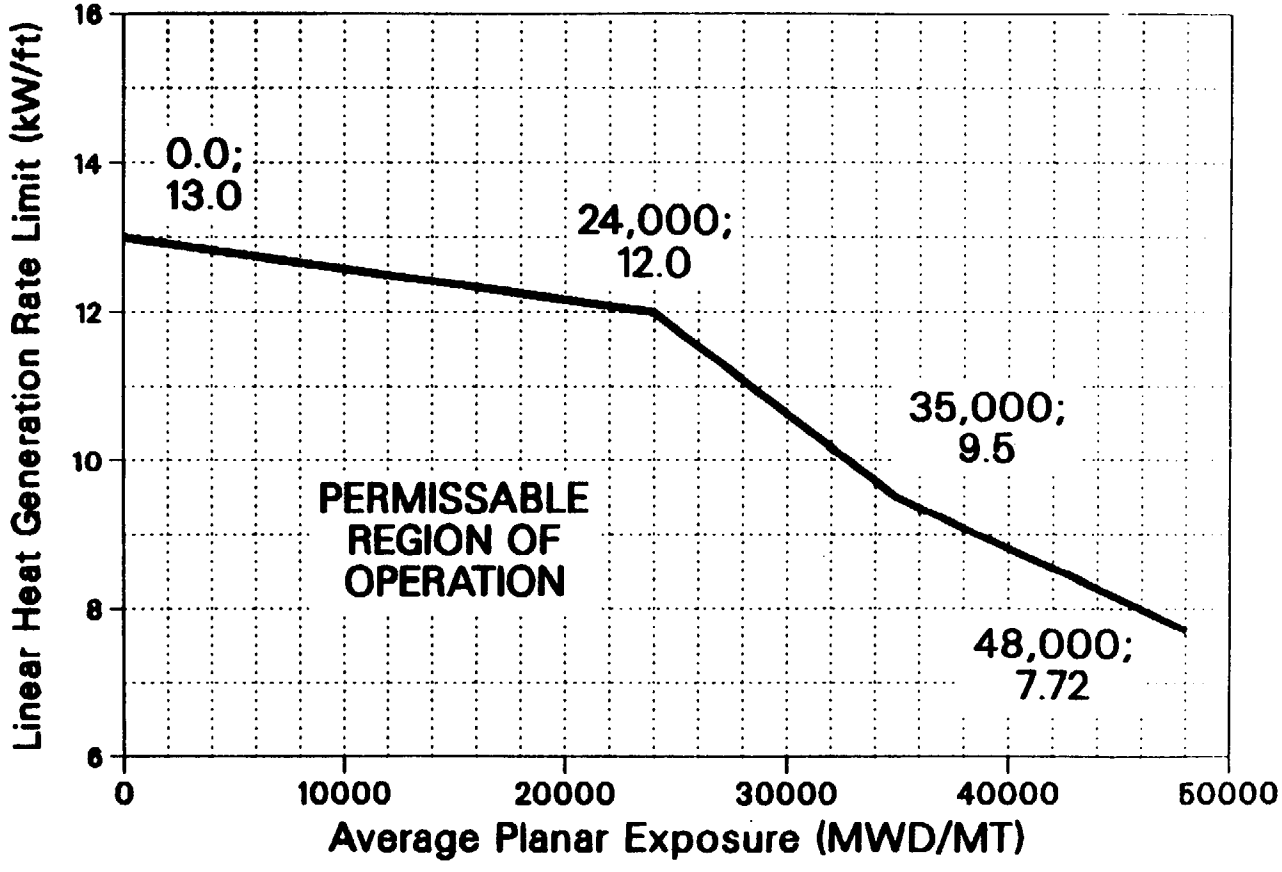
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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



LINEAR HEAT GENERATION RATE (LHGR) LIMIT
VERSUS AVERAGE PLANAR EXPOSURE
ANF 9X9 FUEL
FIGURE 3.2.4-2

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 1 hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 1 hour.

NOTES

* With THERMAL POWER ≥ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

*** Not required when eight or fewer fuel assemblies (adjacent to the SRMs) are in the core.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

b. This function shall be automatically bypassed if detector count rate is ≥ 100 cps or the IRM channels are on range 3 or higher.

c. This function is automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function is automatically bypassed when the IRM channels are on range 3 or higher.

e. This function is automatically bypassed when the IRM channels are on range 1.

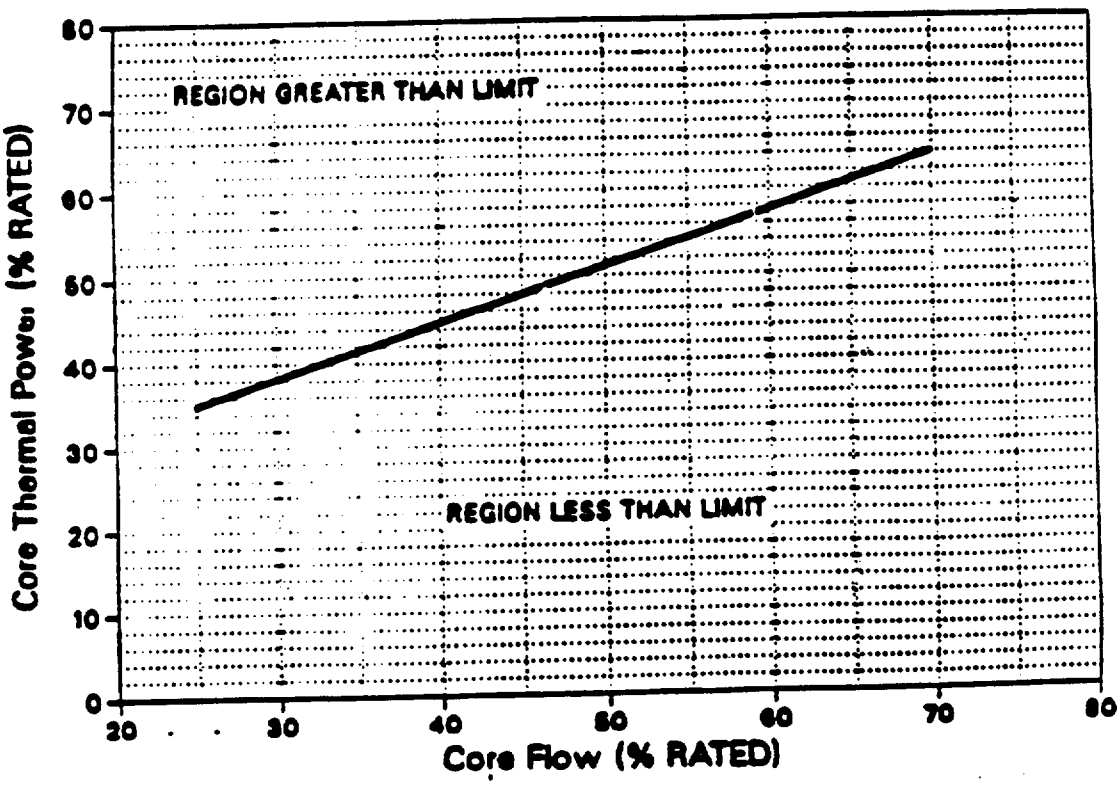
TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale ##	$\leq$ 0.66 W + 42%	$\leq$ 0.66 W + 45%
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5/125 divisions of full scale	$\geq$ 3/125 of divisions full scale
2. APRM		
a. Flow Biased Neutron Flux - Upscale ##	$\leq$ 0.58 W + 50%*	$\leq$ 0.58 W + 53%*
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 2 \times 10^5$ cps	$\leq 4 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 0.7 cps**	$\geq$ 0.5 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 divisions of full scale	$\leq$ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 divisions of full scale	$\geq$ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level - High	$\leq$ 44 gallons	$\leq$ 44 gallons
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 108/125 divisions of full scale	$\leq$ 111/125 divisions of full scale
b. Inoperative	NA	NA
c. Comparator	$\leq$ 10% flow deviation	$\leq$ 11% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.
**Provided signal-to-noise ratio is $\geq$2. Otherwise, 3cps as trip setpoint and 2.8cps for allowable value.
##See Specification 3.4.1.1.2.a for single loop operation requirements.

Figure 3.4.1.1.1-1
THERMAL POWER LIMITATIONS



3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed $\leq$ 80% of the rated pump speed, 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
$\leq$ 0.58W + 54%	$\leq$ 0.58W + 57%.

3. Specification 3.2.1: The MAPLHGR limits shall be as specified in Figures 3.2.1-2 and 3.2.1-3.

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

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

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

a. 1.42,

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.

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

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

b. APRM and LPRM*** neutron flux noise levels shall be less than three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3/4.1.1.1-1.

c. Total core flow shall be greater than or equal to 42 million lbs/hr when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1.1-1.

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

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 Specification 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 Reference 1 or 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 $\geq$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 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

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.

For single loop operation, the MAPLHGR limits are multiplied by a factor of 1.0 for ANF fuel. This multiplication factor is derived from LOCA analyses initiated from single loop operation conditions. The resulting MAPLHGR limits for single loop operation assure the peak cladding temperature 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) and for the Recirculation Pump Seizure Accident.

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.

THERMAL POWER, core flow, and neutron flux noise level limitations are prescribed in accordance with the recommendations of General Electric Service Information Letter No. 380, Revision 1, "BWR Core Thermal Hydraulic Stability," dated February 10, 1984.

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.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 10 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. NPF-14

PENNSYLVANIA POWER AND LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION UNIT 1

1.0 INTRODUCTION

By letter dated February 2, 1989, the Pennsylvania Power and Light Company (the licensee) requested an amendment to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station (SSES), Unit 1. The proposed amendment would support Cycle 5 operation of the SSES Unit 1 with 9x9 reload fuel supplied by Advanced Nuclear Fuels Corporation (ANF).

The SSES Unit 1 Cycle 5 (hereinafter referred to as S1C5) reload will consist of 228 new ANF 9x9 (ANF-4) fuel assemblies, 340 once burned ANF 9x9 (XN-3) fuel assemblies, 292 twice burned ANF 8x8 (XN-2) fuel assemblies, and 4 thrice burned ANF 8x8 (XN-1) fuel assemblies. S1C5 will contain no General Electric Company (GE) fuel assemblies. The new 9x9 bundles are comprised of 79 fuel rods and two water rods. The new 9x9 ANF fuel has similar operating characteristics (thermal-hydraulic and nuclear) to the previously used ANF 9x9 reload fuel. In support of the S1C5 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).

2.0 EVALUATION

2.1 Fuel Mechanical Design

The S1C5 core reload will include 228 ANF 9x9 fuel bundles with the designation ANF-4. These reload bundles contain 79 fuel rods and 2 water rods. The 228 fuel bundles will have a bundle average enrichment of 3.33 weight percent uranium-235. The fuel design and safety analysis are described in the SSES Unit 1 specific report PL-NF-89-001 (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 gives the pertinent design data for ANF 9x9 fuel. Neutronic values specific to the S1C5 reload are given in Table 4.1 of ANF-88-169 (Ref. 4). The burnable poison rods contain 4.0 weight percent gadolinia blended with 3.27 weight percent uranium-235. The ANF-4 fuel bundles are designed to fit into existing GE channel boxes. The analyses for S1C5 support fuel bundle discharge exposures of 35,000 MWd/MTU for ANF 8x8 fuel and

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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 S1C5 reload to be acceptable.

2.2 Rod Pressure

For the S1C5 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.

2.3 Fuel Rod Bow

S1C5 is expected to result in 8x8 and 9x9 ANF fuel bundle exposures less than the assembly maximum allowed burnups. The staff has reviewed Reference 9 which provides additional rod bow measurements on 9x9 Lead Test Assemblies and has concluded that assembly discharge exposures of 40,000 MWd/MTU are acceptable for the ANF 9x9 fuel design (Ref. 10).

2.4 Fuel Centerline Melting

The design basis for the ANF fuel centerline temperature is that no fuel centerline melting should result from normal operation and anticipated operational occurrences. The results of an evaluation reported in the S1C5 reload analysis report were based on the approved RODEX2A code (Ref. 11). The staff has concluded that the generic methodology for the ANF 9x9 fuel is acceptable for the S1C5 reload fuel.

2.5 Linear Heat Generation Rate (LHGR) Limit for ANF 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 (Figure 3.2.4-2 attached to Reference 1). This Figure was approved in Reference 6 to reflect the design values which have been previously 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 LHGR limits for the 9x9 fuel to be acceptable.

2.6 LOCA-Seismic Mechanical Response

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.

2.7 Nuclear Design

The ANF nuclear design methodology for S1C5 is that presented in XN-NF-80-19(A), Volume 1 and Volume 1 Supplements 1 and 2 (Ref. 12), which were reviewed and approved by the staff for generic application to BWR core reloads. The S1C5 licensing analysis employed a new cross section-void history correlation different from that used in Reference 12. The new correlation provides an increased accuracy in cross section determination at exposures greater than 30,000 MWd/MTU. ANF has submitted a letter (Ref. 13) describing this new cross section-void history correlation to the NRC. Use of the correlation by ANF has been approved by the NRC.

The beginning of cycle shutdown margin is calculated to be 1.82 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 4.0 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 1.10349, the ANF calculations show that the staff's acceptance criterion is met for the new fuel storage vault under dry and flooded conditions.

ANF also performed analyses for 9x9 fuel stored in the spent fuel pool. A maximum enriched zone of less than 4.0 weight percent uranium-235 with an uncontrolled, zero void, cold, k-infinity of less than or equal to 1.457 meets the staff acceptance criterion of k-effective no greater than 0.95. Since the ANF-4 9x9 fuel has a zone average enrichment of 3.44 weight percent uranium-235 and a maximum k-infinity of 1.2206 at peak reactivity exposure, the staff's acceptance criterion for spent fuel storage is met for the ANF-4 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.

2.8 Thermal-Hydraulic Design

The review of the thermal-hydraulic aspects of the S1C5 reload consisted of the following: (a) the compatibility of the ANF 9x9 and prior ANF 8x8 fuel assemblies, (b) the fuel cladding integrity safety limit, (c) the bypass flow characteristics, and (d) thermal-hydraulic stability.

The objective of the review was to confirm that the thermal-hydraulic design of the reload core was accomplished using acceptable analytical methods, provided an acceptable margin of safety from conditions which would lead to fuel damage during normal operation and anticipated operational occurrences and ensured that the core is not susceptible to thermal-hydraulic instability.

2.8.1 Hydraulic Compatibility

Since a BWR core is a series of parallel flow channels connected to a common lower and upper plenum, the total pressure drop across the assemblies will be equal. However, differences in the hydraulic resistances of the fuel designs may cause variations in axial pressure drop profiles across the assemblies. Component hydraulic resistances for the constituent fuel types in the S1C5 core have been determined in single phase flow tests of full scale assemblies. Analyses of the effects of hydraulic compatibility on thermal margin have been considered in the S1C5 reload. Based on the staff's review of the information provided in the pertinent documentation and on the fact that the staff has previously approved coresidence of ANF 9x9 and ANF 8x8 fuel for Susquehanna Unit 1, and on the hydraulic equivalence of the ANF 9x9 and ANF 8x8 fuel designs, the staff concludes that the two ANF fuel types are hydraulically compatible.

2.8.2 Minimum Critical Power Ratio Safety Limit

The minimum critical power ratio (MCPR) safety limit for the S1C5 reload was determined by the licensee to be 1.06 for all fuel types. The methodology for S1C5 is based on the ANF critical power methodology in XN-NF-524, Revision 1 (Ref. 14), which has been approved by the staff (Ref. 15). The XN-3 correlation used to develop the MCPR safety limit has been approved for the ANF 9x9 fuel (Ref. 16). The methodology of XN-NF-524, Revision 1 was applied generically for S1C5 and is considered applicable to both ANF 9x9 and ANF 8x8 fuel types. The staff has verified through its review of the S1C5 transient analysis report (Ref. 3) that the methodology for determining uncertainties and the application in determining the MCPR safety limit is in accordance with NRC approved methodology and is acceptable.

2.8.3 Core Bypass Flow

The core bypass flow fraction has been calculated to be 9.9 percent of total core flow using the approved methodology described in XN-NF-524(P)(A), Revision 1 (Ref. 14). This is used in the MCPR safety limit calculations and as input to the S1C5 transient analyses and is acceptable.

2.8.4 Thermal-Hydraulic Stability

The licensee's response to the recently issued NRCB 88-07, Supplement 1 (Ref. 17) is under development. The licensee has not completed preparation of revised Technical Specifications to implement the interim recommendations of Reference 17. The licensee is also an active participant in the BWR Owners Group which is attempting to develop a final resolution of the stability issue.

The licensee performed stability analyses using the COTRAN code. The analysis predicted stable reactor operation at the boundary of the region defined by the area above and to the left of the 80 percent rod line and the 45 percent constant flow line. Operation outside or on the boundary of this region is supported by these COTRAN calculations and result in decay ratios of less than or equal to 0.75 as required by the NRC Safety Evaluation Report (SER) on COTRAN (Ref. 18). This region is the same as previously specified for Susquehanna Unit 1.

The licensee performed stability tests in Susquehanna Unit 2 during initial startup of Cycles 2 and 3. The test results for S2C2 (Ref. 19) show very low decay ratios with a core containing 324 ANF 9x9 fuel assemblies. The ANF "ANNA" software was used to analyze APRM signals from the S2C3 startup and gave a "measured" decay ratio of about 0.37 at a core condition of 60 percent power and 46 percent flow. The S2C3 core contains 556 ANF 9x9 fuel assemblies while the S1C5 core will contain 468 ANF 9x9 fuel assemblies. Therefore, about the same stability behavior is expected for S1C5 as was obtained for S2C3.

Based on the above discussion, we conclude that the licensee has acceptably addressed the thermal-hydraulic stability issue for S1C5. In addition, PP&L has completed and implemented the NRC requested actions specified in Supplement 1 to NRC Bulletin 88-07, "Power Oscillations in Boiling Reactors."

2.9 Transient and Accident Analyses

2.9.1 Operational Transients

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 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 0.27 for ANF 9x9 fuel, which is the most limiting fuel type. The most limiting local transient, the control rod withdrawal error (RWE), was analyzed to support a rod block monitor (RBM) setpoint of 108 percent and resulted in a delta CPR of 0.27. The LRWOB and the RWE events were the most limiting events for S1C5 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 is limiting and MCPR Operating limits for manual flow control reduced flow operation for S1C5 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. 20) 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 S1C5. 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 S1C5.

Compliance with overpressurization criteria was demonstrated by analysis of the main steam isolation valve (MSIV) closure event assuming MSIV position switch scram failure and an MSIV closure time of 2.0 seconds. Six safety-relief valves were assumed to be out-of-service. Maximum pressure was 105 percent of vessel design pressure, well within the 110 percent criterion. The calculation was done with approved methodology and the results are acceptable.

2.9.2 Postulated Accidents

The LOCA analyses for the Susquehanna plants (Ref. 21) was performed for a full core of ANF 9x9 fuel and is applicable for the S1C5 residual and reload ANF fuel. 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 219 cal/gm, which is well below the design limit of 280 cal/gm, and less than 560 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. 22). The staff concludes that the analysis and results for the S1C5 CRDA are acceptable.

2.9.3 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 S1C5 (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. The delta CPR for the SLO pump seizure event was determined to 0.35. Previous analyses reported by

the licensee (Refs. 23 and 24) have shown that other events which could be affected by SLO were non-limiting when analyzed under SLO conditions. SLO for S1C5 must maintain the 80 percent recirculation pump speed restriction because of the previous GE vessel internal vibration analysis, as discussed in Reference 23.

One of the stability tests performed for Susquehanna Unit 2 Cycle 2 was performed under SLO conditions. The measured decay ratio was 0.30 with a standard deviation of 0.064 at core conditions of 55 percent power and 44 percent flow. ANF analysis of this test with the computer code COTRAN gave a calculated decay ratio of 0.29. The licensee states that this data, the stability results presented in Reference 4, and S1C5 Technical Specification stability surveillance requirements support SLO in S1C5.

The original GE SLO analysis required adjustment of the APRM scram, APRM rod block, and the Rod Block Monitor setpoints in SLO to bound the changes in the assumed recirculation drive flow to core flow relationship between two loop operation and SLO. The GE analysis indicated that the two loop to single loop change is less than 7 percent drive flow for a given core flow. The licensee's data for Susquehanna indicate that a value of 8.5 percent would bound differences between two loop operation and SLO. This value of 8.5 percent will be incorporated into the Susquehanna Technical Specifications and is acceptable.

2.10 Technical Specification Changes

The following Technical Specification (TS) changes have been proposed for operation of S1C5:

1. Specification 2.2.1 Reactor Protection System Instrumentation Setpoints

   The change to this Specification is to Functional Unit 2.b of Table 2.2.1-1, where a footnote has been added to the allowable value. This change is an editorial change and is, therefore, acceptable.

2. Basis 2.0 Introduction

   Editorial changes are made to this Basis. The references to GE and EXXON fuel are deleted and replaced with a reference to ANF fuel. S1C5 will not contain GE fuel and ANF has replaced EXXON as a corporate entity. Revision 1 is appended after report XN-NF-524(A). These changes are consistent with the proposed changes and are, therefore, acceptable.

3. Basis 2.1.1 Thermal Power, Low Pressure or Low Flow

   The reference to GE fuel is deleted since S1C5 will not contain GE fuel. This is consistent with the proposed changes and is, therefore, acceptable.

4. Basis 2.1.2 Thermal Power, High Pressure and High Flow

   This change involves adding Revision 1 to report XN-NF-524(A). This is an editorial change and is, therefore, acceptable.

5. Specification 3/4.2.1 Average Planar Linear Heat Generation Rate

The references to GE fuel and the MAPLHGR limits are deleted because S1C5 will not contain GE fuel. This change is consistent with the proposed changes and is, therefore, acceptable.

6. Specification 3/4.2.2 APRM Setpoints

The reference to GE fuel is deleted because S1C5 will not contain any GE fuel. This change is consistent with the proposed changes and is, therefore, acceptable.

7. Specification 3/4.2.3 Minimum Critical Power Ratio

Operating limit MCPRs have been revised to reflect the results of the cycle specific transient analyses. The methodology used to evaluate the limiting transients and accidents is consistent with previously approved methods and meets all the appropriate criteria. Therefore, the revised MCPRs are acceptable for S1C5 as discussed in this SER.

8. Specification 3/4.2.4 Linear Heat Generation Rate

The limit and reference to GE fuel are being deleted because S1C5 will not contain any GE fuel. These changes are consistent with the proposed changes and are, therefore, acceptable.

9. Specification 3/4.3.6 Control Rod Block Instrumentation

This change corrects an administrative error by applying a footnote to trip functions 1a and 2a in Table 3.3.6-2. The footnote refers to Specification 3/4.1.1.2.a for single loop operation. This change is editorial and is, therefore, acceptable.

10. Specification 3.4.1.1.2 Single Loop Operation

These changes are required to support single loop operation (SLO) for S1C5. Specifications 3.4.1.1.2a.2, 4 and 6 incorporate setpoint adjustments to account for the change in the previous GE value of 7 percent less drive flow for a given core flow, when going from two-loop to SLO, to a new value of 8.5 percent determined by PP&L. Specification 3.4.1.1.2.a.3 is revised to delete reference to GE fuel because S1C5 will not contain GE fuel. New Specification 3.4.1.1.2.a.5 is needed to provide new MCPR limits for ANF fuel. These changes are acceptable because they propose new limits for SLO for S1C5 based on approved methodologies and results.

11. Basis 3/4.2.1 Average Planar Linear Heat Generation Rate

The reference to GE fuel is deleted. Other minor editorial changes are also made. These changes are acceptable.

12. Basis 3/4.2.2 APRM Setpoints

The reference to GE fuel is deleted. This change is acceptable because S1C5 will not contain any GE fuel.

13. Basis 3/4.2.3 Minimum Critical Power Ratio

A reference to EXXON is changed to a reference to ANF. This change is acceptable.

14. Basis 3/4.4.1 Recirculation System

The Basis is changed to reflect the MAPLHGR multiplier for ANF fuel for SLO. The Basis is also changed by replacing the 7 percent for the decrease in recirculation drive flow for a given core flow by the new PP&L determined value of 8.5 percent. These changes are acceptable because the analysis for SLO for S1C5 was performed using approved methodologies which gave acceptable results.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes 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/or 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 13767) on April 5, 1989 and consulted with the State of Pennsylvania. No public comments were received, and the State 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 the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

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Dated: May 15, 1989

5.0 REFERENCES

1. Letter (PLA-3141) from H.W. Keiser (PP&L) to W.R. Butler (NRC), "Proposed Amendment 119 to License No. NPF-14: Unit 1 Cycle 5 Reload," dated February 2, 1989.

2. PL-NF-89-001, "Susquehanna SES Unit 1-Cycle 5: Reload Summary Report," dated January 1989.

3. ANF-88-168, "Susquehanna Unit 1 Cycle 5: Plant Transient Analysis," dated January 1989.

4. ANF-88-169, "Susquehanna Unit 1 Cycle 5 Reload Analysis: Design and Safety Analyses," dated January 1989.

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,'" dated 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 BWR Fuel," dated 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)," dated June 24, 1986.

9. Letter from G.N. Wood (ANF) to G.C. Lainas (NRC), "Additional Information on Rod Bowing," dated March 11, 1987.

10. Letter from A.C. Thadani (NRC) to D.A. Adkisson (ANF), "Acceptance for Referencing of Licensing Topical Report XN-NF-82-06, Supplement 1, Revision 2."

11. 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,'" dated June 24, 1986.

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," dated March 1983.

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

14. XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for BWRs," dated November 1983.

15. Letter from C.O. Thomas (NRC) to J.C. Chandler (ANF), "Acceptance for Referencing of Licensing Topical Report XN-NF-524(P)," dated October 31, 1983.

16. Letter from C.O. Thomas (NRC) to J.C. Chandler (ANF), "Acceptance for Referencing of Licensing Topical Report XN-NF-734, 'Confirmation of the XN-3 Critical Power Correlation for 9x9 Fuel Assemblies,'" dated February 1, 1985.

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

18. XN-NF-691(P)(A) and Supplement 1, "Stability Evaluation of Boiling Water Reactor Cores - Sensitivity Analyses and Benchmark Analysis," dated August 1984.

19. XN-NF-86-90, Supplement 1, "Susquehanna Unit 2 Cycle 2 Stability Test Results," dated January 1987.

20. 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," dated February 1987.

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

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

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

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