April 25, 1988

Docket No. 50-388

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BClayton	RBlough

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

Dear Mr. Keiser:

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO SUPPORT CYCLE 3 OPERATION (TAC NO. 66921)

RE: SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

The Commission has issued the enclosed Amendment No.45 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. This amendment is in response to your letter dated December 23, 1987.

This amendment changes SSES, Unit 2 Technical Specifications to support the fuel reload and Cycle 3 operation.

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. 45 to
   License No. NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

PDI-2/LA	PDI-2/PM	PDI-2/D	OGC
MO'Brien	MThadani:mr	WButler	
4/13/88	4/6/88	4/21/88	4/5/88

3. This license amendment is effective prior to startup for Cycle 3 operation.

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: April 25, 1988

PDI-2/LA
MO'Brien
4/ /88

PDI-2/PM
MThadani:mr
4/6/88

OGC
4/8/88

PDI-2/D
WButler
4/21/88

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

April 25, 1988

Docket No. 50-388

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Vice President
Nuclear Operations
Pennsylvania Power and Light Company
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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. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. NPF-22

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated December 23, 1987 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-22 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 45 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 prior to startup for Cycle 3 operation.

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: April 25, 1988

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	MThadani:mr		WButler
/ /88	4/6/88	4/8/88	4/21/88

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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: April 25, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following pages of the Appendix A Technical Specifications with enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The overleaf pages are provided to maintain document completeness.*

REMOVE	INSERT
iii	iii*
iv	iv
xi	xi*
xii	xii
xxi	xxi*
xxii	xxii
B 2-1	B 2-1
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 2-6	3/4 2-6*
3/4 2-6a	3/4 2-6a
-	-
3/4 2-7	3/4 2-7*
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
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 3-53	3/4 3-53*
3/4 3-54	3/4 3-54
3/4 4-1	3/4 4-1
3/4 4-1a	3/4 4-1a*
3/4 4-1b	3/4 4-1b
3/4 4-1c	3/4 4-1c

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

REMOVE	INSERT
3/4 4-1d	3/4 4-1d
3/4 4-1e	3/4 4-1e*
3/4 4-1f	3/4 4-1f*
-	3/4 4-1g
3/4 4-2	3/4 4-2*
-	-
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2*

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ADMINISTRATIVE CONTROLS

LIST OF FIGURES

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 Specification 2.1.2 for both GE and 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)).

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 9 x 9 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For the ANF and GE 8 x 8 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 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 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 PLANAR EXPOSURE for GE fuel and AVERAGE BUNDLE EXPOSURE for ANF fuel shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3.*

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, 3.2.1-2, or 3.2.1-3, 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, 3.2.1-2, and 3.2.1-3:

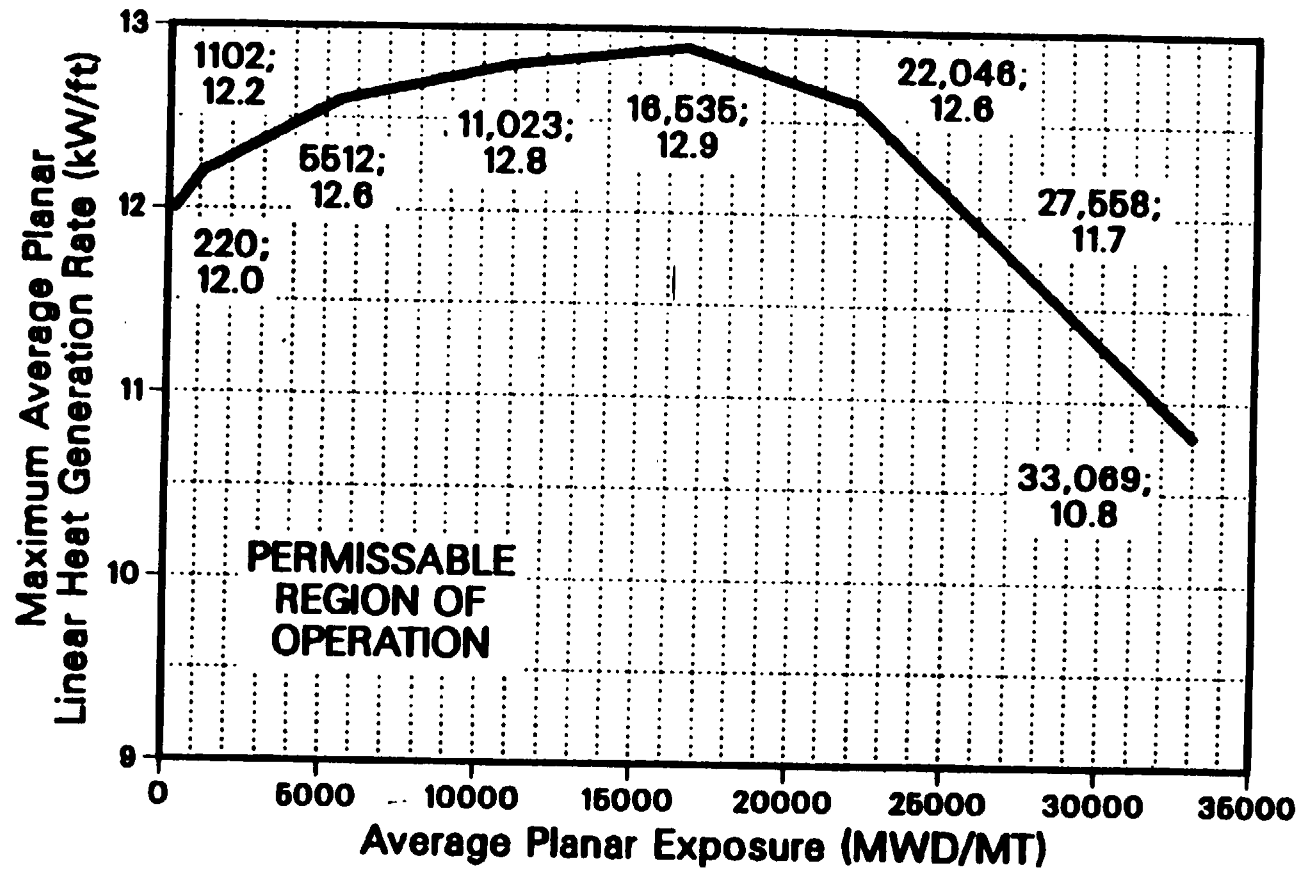
a. At least once per 24 hours,

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.

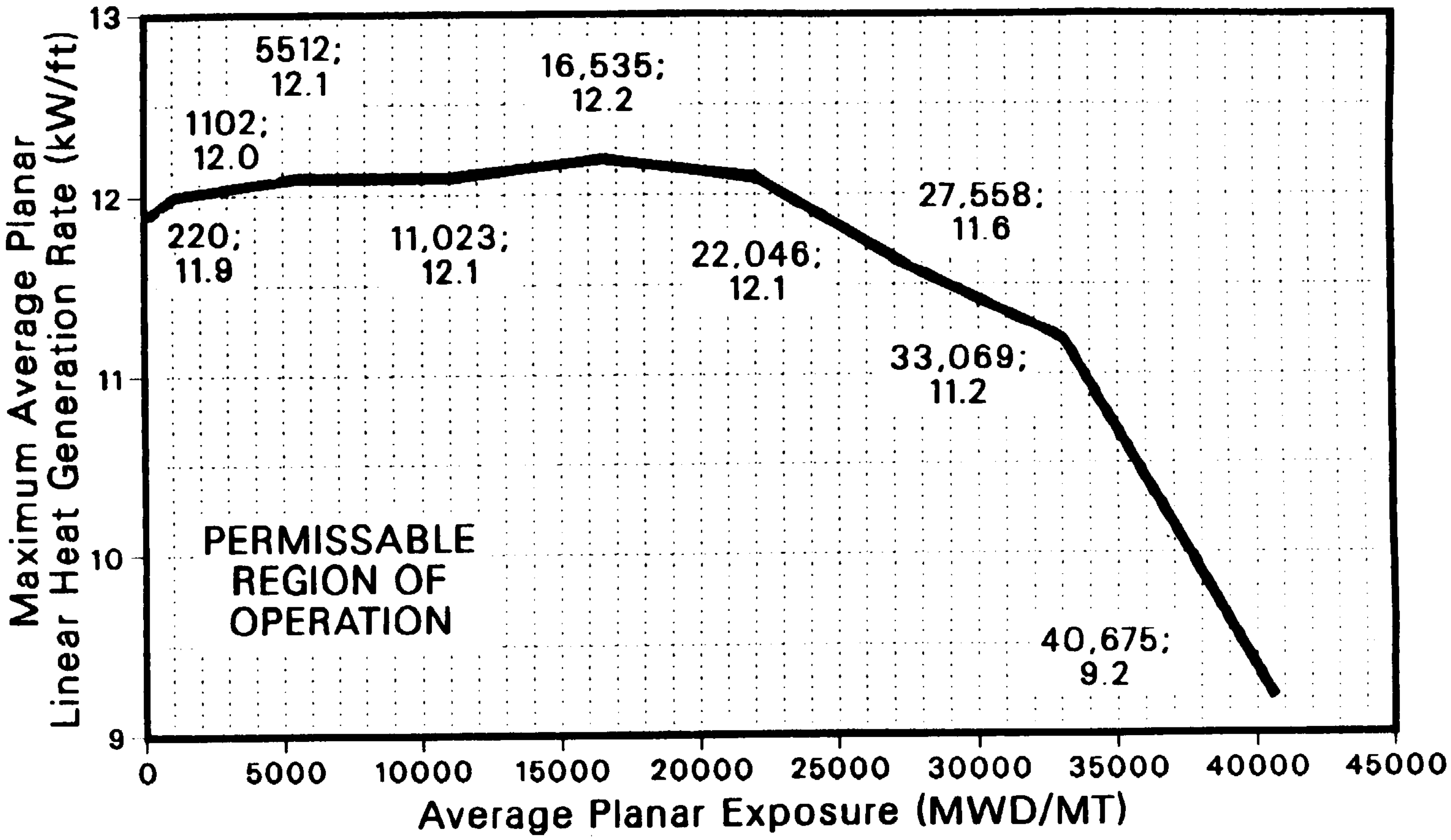
*See Specification 3.4.1.1.2.a for single loop operation requirements.



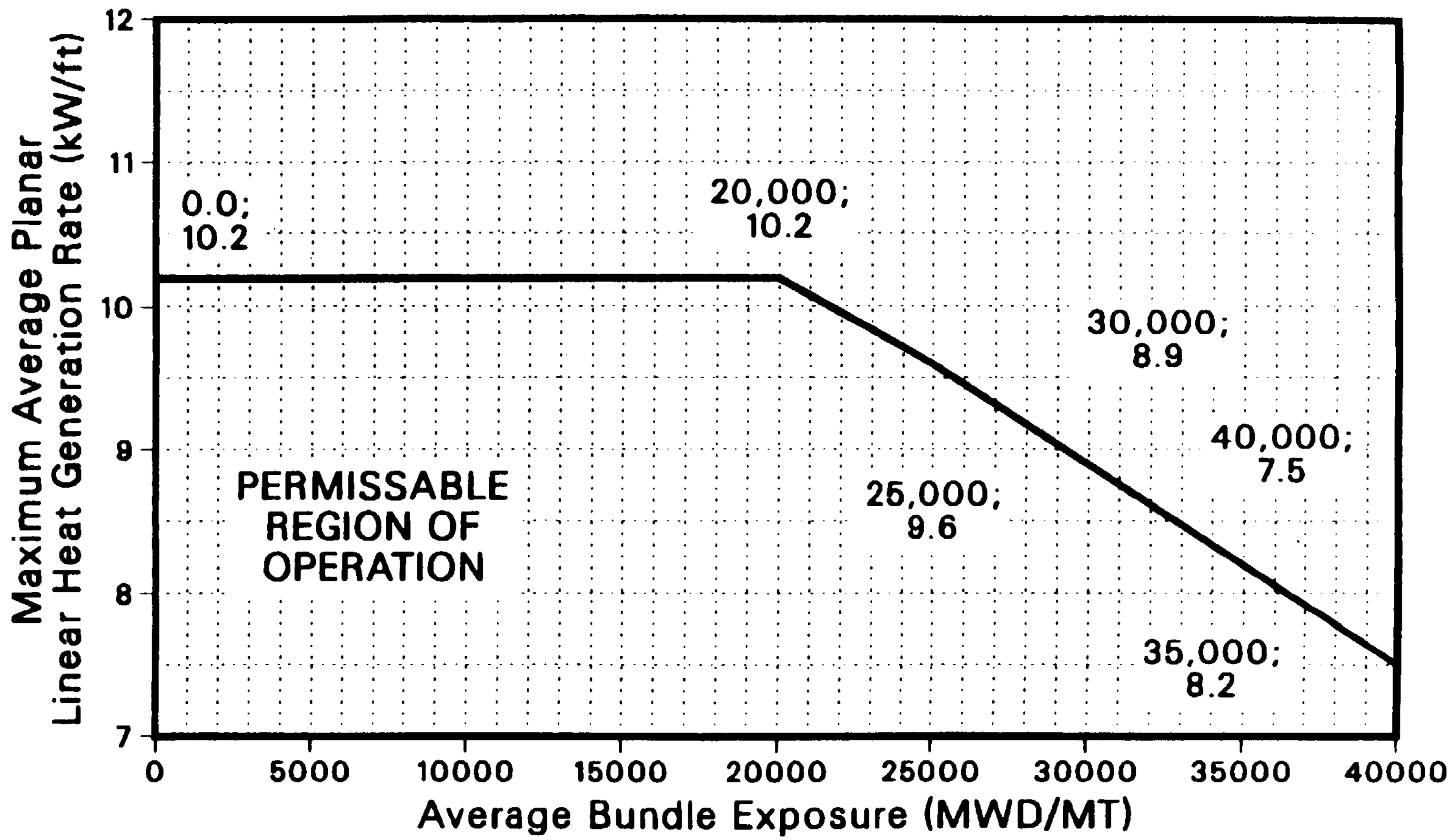
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE GE FUEL TYPES 8CR183 (1.83% ENRICHED)

FIGURE 3.2.1-1

following the unit's first refueling outage
effective upon startup



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
GE FUEL TYPES 8CR233 (2.33% ENRICHED)
FIGURE 3.2.1-2



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
ANF 9X9 FUEL
FIGURE 3.2.1-3

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. Where:

a. The FRACTION OF LIMITING POWER DENSITY (FLPD) for GE fuel is the actual LINEAR HEAT GENERATION RATE (LHGR) divided by 13.4 per Specification 3.2.4.1, and

b. 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.

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:

*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.

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

4.2.2 (Continued)

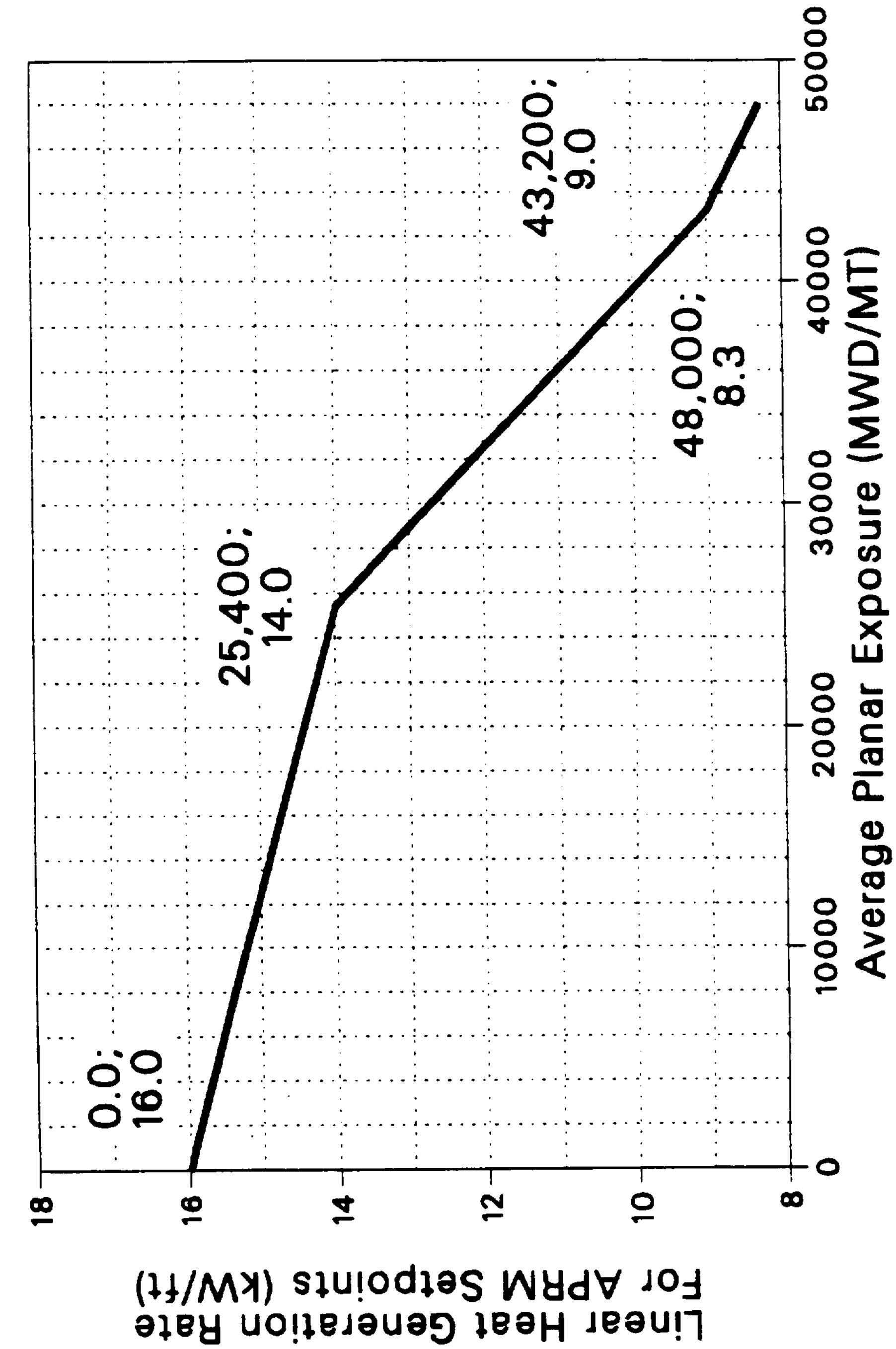
a. At least once per 24 hours,

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.

effective upon startup
following the unit 2 first refueling outage



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:

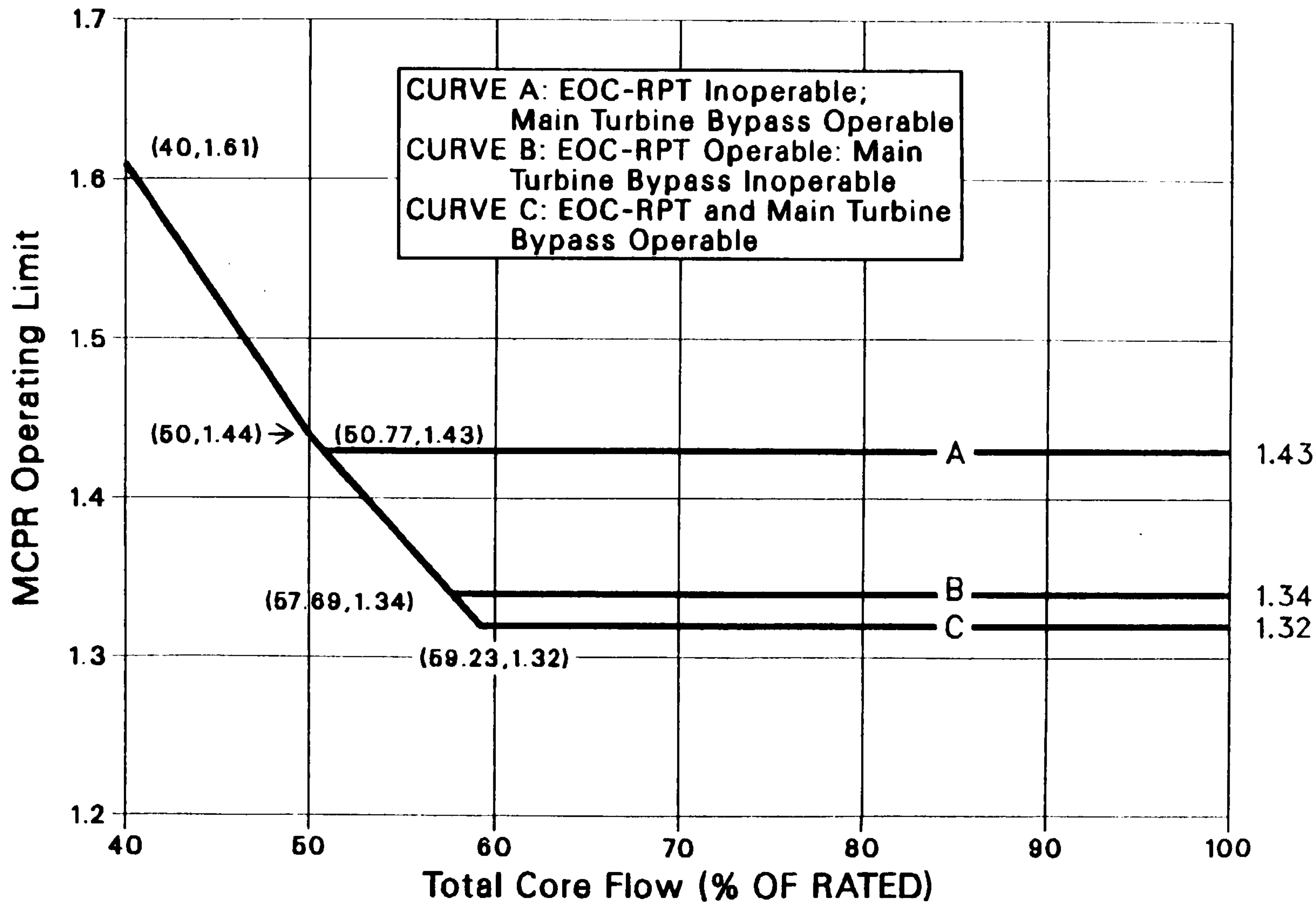
a. At least once per 24 hours,

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

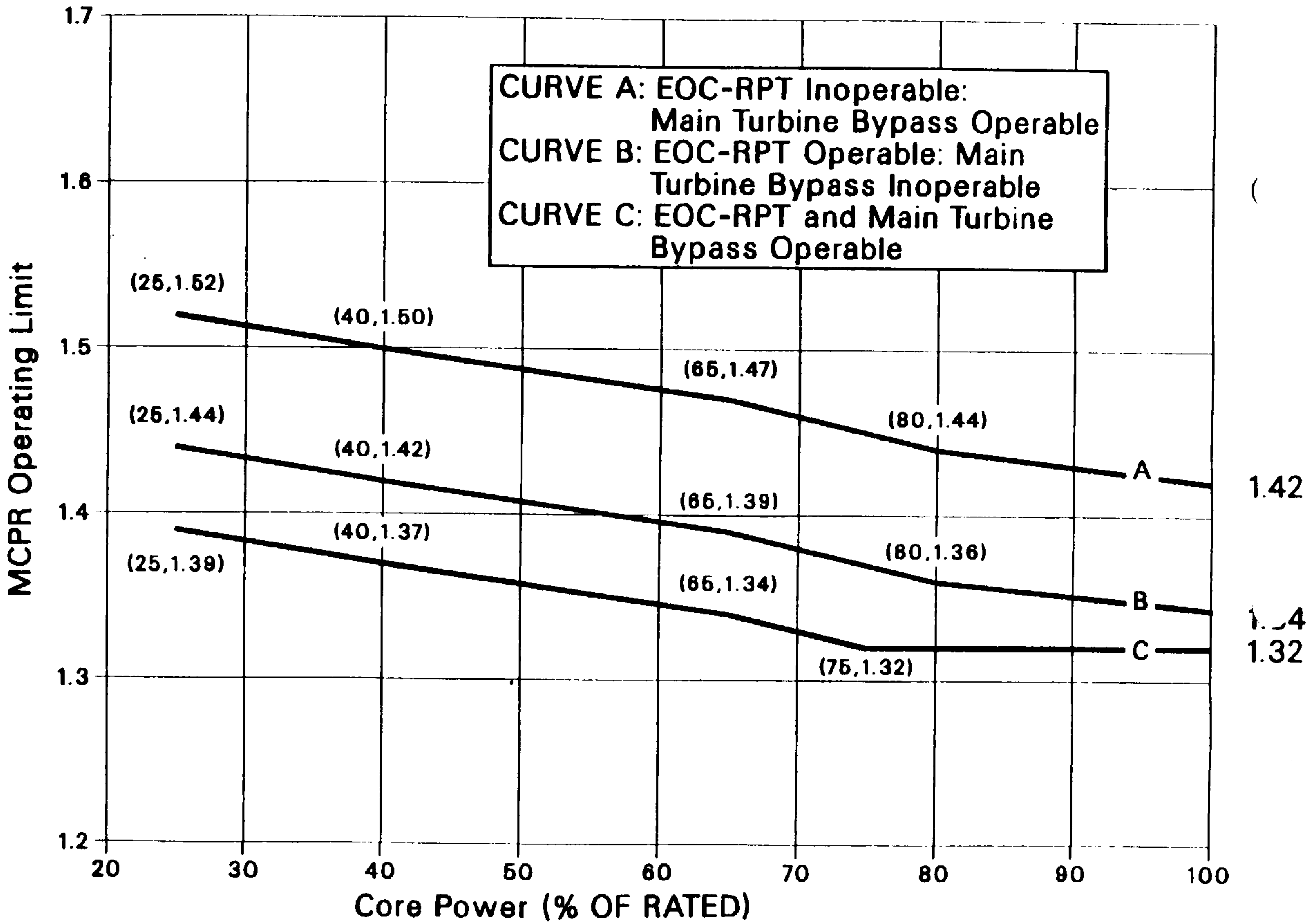
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

effective upon startup following the unit 2 first refueling outage



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

GE FUEL

LIMITING CONDITION FOR OPERATION

3.2.4.1 The LINEAR HEAT GENERATION RATE (LHGR) for GE fuel shall not exceed 13.4 kw/ft.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, 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

4.2.4.1 LHGRs for GE 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.

effective upon startup following the unit 2 first refueling outage

3/4.2.4 LINEAR HEAT GENERATION RATE

ANF FUEL

LIMITING CONDITION FOR OPERATION

3.2.4.2 The LINEAR HEAT GENERATION RATE (LHGR) for ANF fuel shall not exceed the LHGR limit determined from Figure 3.2.4.2-1.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, 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

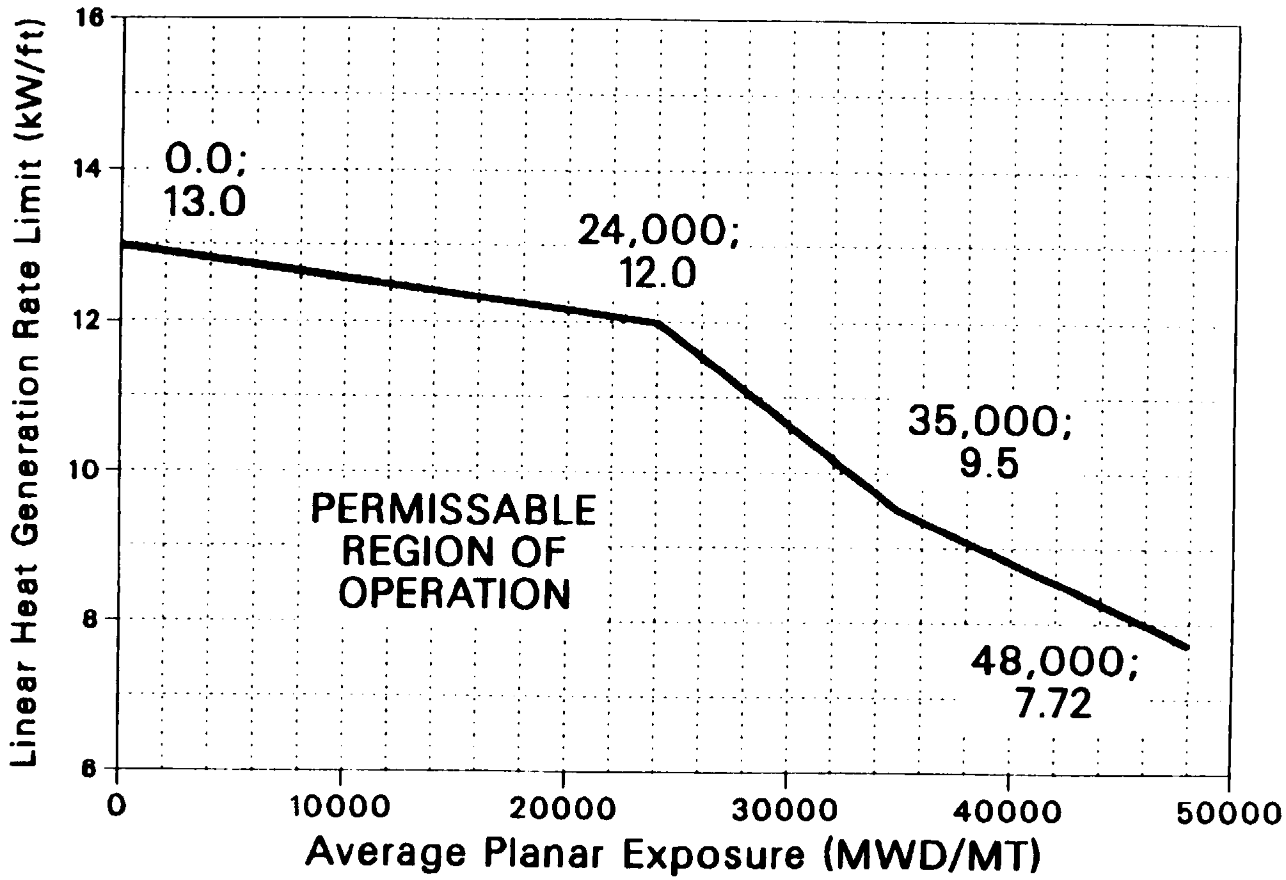
4.2.4.2 LHGRs for ANF fuel shall be determined to be equal to or less than the limit:

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 9X9 FUEL
FIGURE 3.2.4.2-1

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 $\geq$ 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 of 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 $\geq$ 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, 3 cps as trip setpoint and 2.8 cps for allowable value.

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

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS - TWO LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.1 Two reactor coolant system recirculation loops shall be in operation and:

a. Total core flow shall be greater than or equal to 55 million lbs/hr, or

b. The reactor is at a THERMAL POWER/core flow condition less than or equal to the limit specified in Figure 3.4.1.1.1-1.

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

ACTION:

a. With one reactor coolant system recirculation loop not in operation, comply with the requirements of Specification 3.4.1.1.2, or take the associated ACTION.

b. With no reactor coolant system recirculation loops in operation, immediately initiate an orderly reduction of THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1.1-1, and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 55 million lbs/hr and the reactor at a THERMAL POWER/core flow condition greater than the limit specified in Figure 3.4.1.1.1-1:

   1. Restore the reactor to a THERMAL POWER/core flow condition less than or equal to the limit specified in Figure 3.4.1.1.1-1, or

   2. Increase core flow to greater than 55 million lbs/hr, or

   3. Determine the APRM and LPRM*** neutron flux noise levels within 1 hour, and:

      a) If the APRM and LPRM*** neutron flux noise levels are less than three times their established baseline levels, continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER, or

      b) If the APRM or LPRM*** neutron flux noise levels are greater than or equal to three times their established baseline levels, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 55 million lbs/hr, and/or by returning the reactor to a THERMAL POWER/core flow condition less than or equal to the limit specified in Figure 3.4.1.1.1-1.

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*See Special Test Exception 3.10.4.

***Detectors A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

#See Specification 3.4.1.1.2 for single loop operation requirements.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1.1 Each pump discharge valve and bypass valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.1.2 Each pump discharge bypass valve, if not OPERABLE, shall be verified to be closed at least once per 31 days.

4.4.1.1.1.3 Each pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 102.5 and 105%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.1.4 Establish a baseline APRM and LPRM neutron flux noise value at a point within 5% RATED THERMAL POWER of the 100% rated rod line with total core flow between 35% and 50% of rated total core flow during startup testing following each refueling outage.

**If not performed within the previous 31 days.

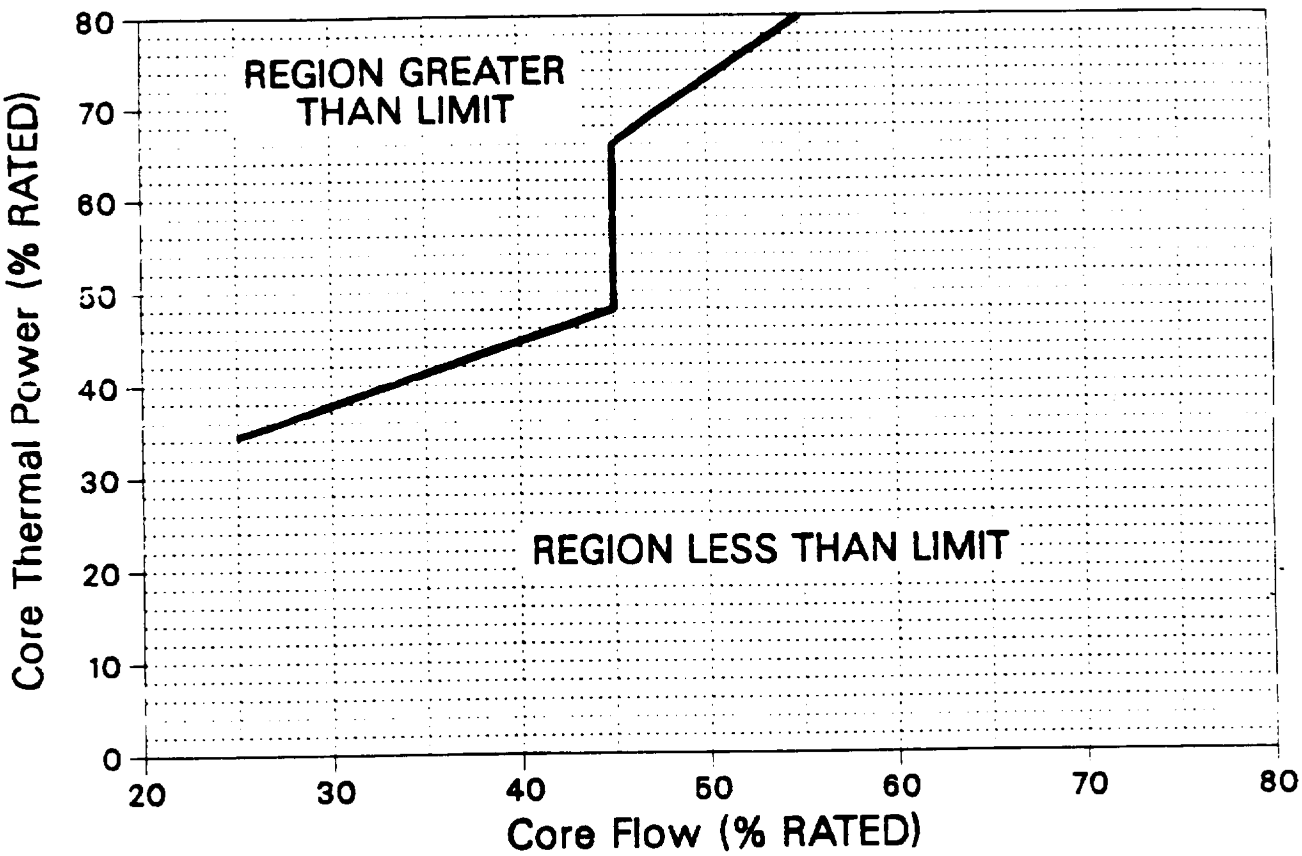

Figure 3.4.1.1.1-1
THERMAL POWER/CORE FLOW LIMITATIONS

LIMITING CONDITION FOR OPERATION

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 + 55%	$\leq$ 0.58W + 58%.

3. Specification 3.2.1: The MAPLHGR limits shall be the limits specified in Figures 3.2.1-1 and 3.2.1-2 multiplied by 0.81 and Figure 3.2.1-3 multiplied by 1.0.

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

Trip Setpoint	Allowable Value
$S \leq (0.58W + 55\%)T$	$S \leq (0.58W + 58\%)T$
$S_{RB} \leq (0.58W + 46\%)T$	$S_{RB} \leq (0.58W + 49\%)T$

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. 1.37,

   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 + 37%	$\leq$ 0.66W + 40%
b. APRM-Flow Biased	$\leq$ 0.58W + 46%	$\leq$ 0.58W + 49%

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 3/4.1.1.2-1.

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 3.4.1.1.2-1.

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

ACTION:

a. With no reactor coolant system recirculation loops in operation, take the ACTION required by Specification 3.4.1.1.1.

LIMITING CONDITION FOR OPERATION (Continued)

b. With any of the limits specified in 3/4.1.1.2a not satisfied:

1. Upon entering single loop operation, comply with the new limits within 6 hours or be in at least HOT SHUTDOWN within the following 6 hours.

2. If the provisions of ACTION b.1 do not apply, take the ACTION(s) required by the referenced Specification(s).

c. With the APRM or LPRM*** neutron flux noise levels greater than or equal to three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1.2-1, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by initiating an orderly reduction of THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1.2-1. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

d. With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

e. With total core flow less than 42 million lbs/hr when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1.2-1, immediately initiate corrective action by either:

1. Reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1.2-1 within 4 hours, or

2. Increasing total core flow to greater than or equal to 42 million lbs/hr within 4 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.2.1 Upon entering single loop operation and at least once per 24 hours thereafter, verify that the pump speed in the operating loop is $\leq$ 80% of the rated pump speed.

4.4.1.1.2.2 With THERMAL POWER greater than the limit specified in Figure 3.4.1.1.2-1, determine the APRM and LPRM*** neutron flux noise levels within 1 hour. Continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion of the THERMAL POWER increase $\geq$ 5% of RATED THERMAL POWER.

4.4.1.1.2.3 Within 15 minutes prior to either THERMAL POWER increase resulting from a control rod withdrawal or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is $<$ 30%**** of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is $\leq$ 50%**** of rated loop flow:

a. $\leq$ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b.## $\leq$ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c.## $\leq$ 50°F between the reactor coolant within the loop not in operation and operating loop.

4.4.1.1.2.4 a. Establish a baseline APRM and LPRM neutron flux noise value at a point within 5% RATED THERMAL POWER of the 100% rated rod line with total core flow between 35% and 50% of rated total core flow during startup testing following each refueling outage, or

b. In lieu of establishing a single loop operation baseline value, utilize the value established pursuant to Specification 4.4.1.1.1.4 if a baseline value is needed to meet the requirements of Specification 3.4.1.1.2.

4.4.1.1.2.5 The pump discharge valve and bypass valve in both loops shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2.6 The pump discharge bypass valve in the OPERABLE loop, if not OPERABLE, shall be verified to be closed at least once per 31 days.

4.4.1.1.2.7 The pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 102.5% and 105%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.2.8 The pump discharge valve and bypass valve in the inoperable loop, if not OPERABLE, shall be verified to be closed at least once per 31 days.

4.4.1.1.2.9 During single recirculation loop operation, all jet pumps, including those in the inoperable loop, shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur:###

a. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established single recirculation pump speed-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value from single recirculation loop flow measurements.

SURVEILLANCE REQUIREMENTS (Continued)

c. The indicated diffuser -to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop patterns by more than 10%.

4.4.1.1.2.10 The SURVEILLANCE REQUIREMENTS associated with the specifications referenced in 3.4.1.1.2a shall be followed.

* See Special Test Exception 3.10.4.

** If not performed within the previous 31 days.

*** Detectors A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

**** Initial value. Final value to be determined based on startup testing. Any required change to this value shall be submitted to the Commission within 90 days of test completion.

\# See Specification 3.4.1.1.1 for two loop operation requirements.

\## This requirement does not apply when the loop not in operation is isolated from the reactor pressure vessel.

\### During startup testing following each refueling outage, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the performance of subsequent required surveillances.

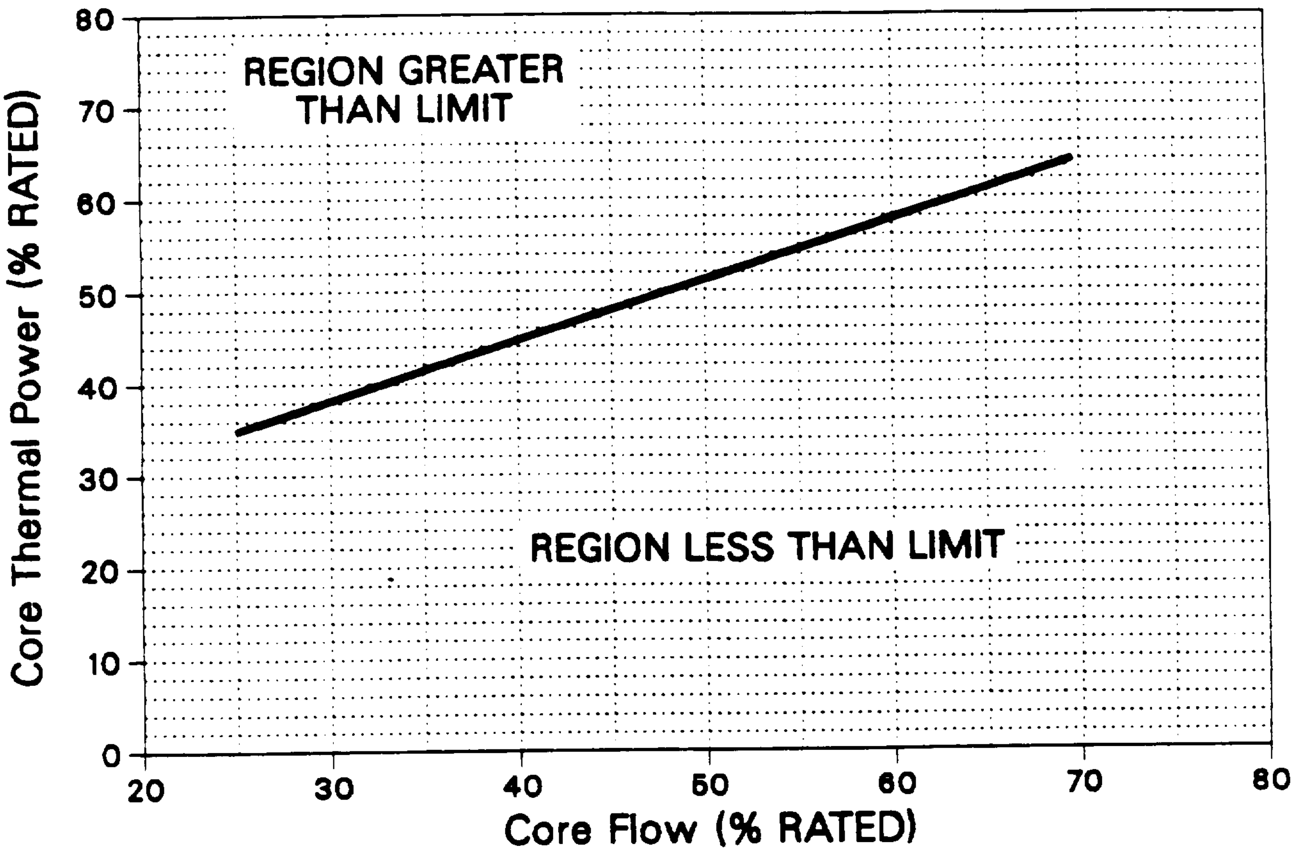

Figure 3.4.1.1.2-1
SINGLE LOOP OPERATION
THERMAL POWER LIMITATIONS

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 when both recirculation loops are in operation.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2** Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours* by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when the recirculation pumps are operating at the same speed:

a. The indicated recirculation loop flow differs by more than 10% from the established pump speed-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established patterns by more than 10%.

*During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.

**See Specification 4.4.1.1.2.9 for single loop operation requirements.

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. For GE fuel, the peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for GE fuel is this LHGR of the highest powered rod divided by its local peaking factor which results in a calculated LOCA PCT much less than 2200°F. 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, 3.2.1-2, and 3.2.1-3.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 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 ≥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 AOO's.

APRM SETPOINTS (Continued)

For GE fuel the T factor used to adjust the APRM setpoints is based on the FLPD calculated by dividing the actual LHGR by the LHGR limit specified for GE fuel in Specification 3.2.4.1.

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 Figure 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 violated 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 violated in the event of a feedwater controller failure initiated from a reduced power condition.

Cycle specific analyses are performed for the most limiting local 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.

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 0.81 for GE fuel and 1.0 for the ANF fuel. These multiplication factors are 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 7% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

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

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. 45 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

1.0 INTRODUCTION

By letter dated December 23, 1987, Pennsylvania Power & Light Company (PP&L) requested an amendment to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. The proposed amendment would change the Technical Specifications to support authorization for Susquehanna Steam Electric Station (SSES) Unit 2 operation with 9X9 Cycle 3 (S2C3) reload fuel supplied by Advanced Nuclear Fuels (ANF) Corporation.

The Susquehanna 2 S2C3 reload will consist of 236 fresh ANF fuel assemblies (XN-2) intermixed with 324 ANF 9X9 assemblies (XN-1) and 204 initial core General Electric (GE) P8X8R assemblies. In support of the S2C3 reload, PP&L submitted topical reports which describe the reload scope, the proposed startup physics tests, the plant transient analysis, and the design and safety analyses.

2.0 EVALUATION

2.1 Fuel Mechanical Design

The S2C3 core reload will include 236 ANF 9X9 fuel assemblies with the designation XN-2. These reload assemblies contain 79 fuel rods and two water rods. The XN-2 reload fuel consists of 140 assemblies which contain nine burnable poison rods and 96 assemblies which contain 10 burnable poison rods. These 236 assemblies will have an assembly average enrichment of 3.33 weight percent (w/o) U-235. The fuel design and safety analysis for the XN-2 fuel are the same as those for the previous cycle XN-1 fuel and are described in the Susquehanna 2 specific report ANF-87-126 and the generic mechanical design report XN-NF-85-67(P)(A), Revision 1. The staff has approved the latter report and issued an SER on July 23, 1986.

Table 2.1 of XN-NF-85-67, Revision 1 gives the pertinent data for the ANF 9X9 fuel. Neutronic values specific to the S2C3 reload are given in Table 4.1 of ANF-87-126, Revision 1. The ANF XN-2 fuel is designed to fit into the existing GE channel boxes. Based on the staff's review of the information presented, the mechanical design of the ANF 9X9 fuel for the S2C3 reload is acceptable.

2.2 Rod Pressure

For the S2C3 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. 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

S2C3 is expected to result in a peak XN-1 assembly exposure of less than 30,000 MWD/MTU at end-of-cycle. 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 XN-1 and XN-2 fuel designs.

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 including anticipated transient occurrences. The results of an evaluation reported in the S2C3 reload analysis were based on the approved RODEX2A code and the staff has concluded that the generic methodology for the ANF 9X9 fuel is acceptable for the S2C3 reload fuel.

2.5 Linear Heat Generation Rate (LHGR) Limit for ANF 9X9 fuel

A figure of LHGR limit versus Planar Exposure (MWD/MT) for the ANF 9X9 fuel type is incorporated into the Susquehanna 2 Technical Specifications. This Figure was approved in the staff's safety evaluation dated July 23, 1986 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, Revision 1. Based on the results of the generic review the staff finds the LHGR limits for the 9X9 fuel acceptable.

3.0 NUCLEAR DESIGN

The ANF nuclear design methodology for S2C3 is that presented in XN-NF-80-19(A), Volume 1, and its Supplements 1 and 2, which were reviewed and approved by the staff for generic application to BWR reloads.

The beginning-of-cycle shutdown margin is calculated to be 1.50 percent delta k/k, and the R factor is zero. Thus the cycle minimum shutdown margin is well in excess of 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 assembly. Based on ANF calculations of 9X9 fuel, an average enrichment of less than 4.0 w/o U-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 maximum enrichment of the new fuel is 3.42 w/o U-235 and the maximum cold, uncontrolled, k-infinity is 1.10349, the calculations show the staff's acceptance criterion is met for the new fuel storage vault under all normal and postulated abnormal conditions.

ANF also performed analyses for 9X9 fuel stored in the SSES spent fuel pool. A maximum enriched zone of less than 4.0 w/o U-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 XN-2 9X9 fuel has an enrichment of 3.44 w/o U-235 and a maximum k-infinity of 1.2206 at peak reactivity, the staff's acceptance criterion for spent fuel storage is also met for ANF 9X9 fuel.

Susquehanna will continue to use the ANF POWERPLEX core monitoring system to monitor reactor parameters. The system has been in use since the previous operating cycle and during Unit 1 Cycles 2, 3, and 4 and has provided suitable monitoring and predictive results.

4.0 THERMAL-HYDRAULIC DESIGN

The review of the thermal-hydraulic aspects of the S2C3 reload consisted of the following: (a) the compatibility of the ANF 9X9 and prior GE 8X8 fuel assemblies; (b) the fuel cladding integrity safety limit; (c) the bypass flow characteristics; (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.

4.1 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 proposed constituent fuel types in the S2C3 core have been determined in single phase flow tests of full scale assemblies. Additional analyses of the effects of hydraulic compatibility on thermal margin were presented in the S2C3 reload reports. 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 GE P8X8R and ANF 9X9 fuel for Unit 2, and on the hydraulic equivalence of the XN-2 9X9 design and the XN-1 9X9 design, the staff concludes that the ANF and GE fuel types in S2C3 are hydraulically compatible.

4.2 Minimum Critical Power Ratio Safety Limit

The minimum critical power ratio (MCPR) safety limit for the Cycle 3 reload was determined by the licensee to be 1.06 for all fuel types. The methodology for Cycle 3 is based on the ANF critical power methodology in XN-NF-524, Revision 1, 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. The methodology of XN-NF-524, Revision 1 was applied generically for the upcoming Cycle 3 and is considered applicable to the resident GE 8X8 fuel as well as the ANF fuel. The staff has verified through its review of the S2C3 transient analysis report 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.

4.3 Core Bypass Flow

The core bypass flow fraction has been calculated as 10.1% of total core flow using the approved methodology described in XN-NF-524(A), Revision 1. This is used in the MCPR safety limit calculation and as input to the Cycle 3 transient analyses and is acceptable.

4.2 Thermal-Hydraulic Stability

The thermal-hydraulic stability of the SSES 2 core was analyzed using the methods identified in Exxon Reports related to Nutronic design and analysis methods, and stability evaluation methods. The licensee has also performed a stability startup test in Unit 2 during initial startup of Cycle 2 to demonstrate stable reactor operation with ANF 9X9 fuel. In addition, the Unit 2 Technical Specifications have implemented surveillances for detecting and suppressing power oscillations. The acceptability of these surveillance requirements as well as the tests and analyses mentioned above have been evaluated for S2C3 in a separate safety evaluation for Unit 2 single loop operation.

5.0 TRANSIENT AND ACCIDENT ANALYSES

5.1 Operational Transients

Various operational transients could reduce the MCPR below the intended safety limit. The most limiting transients have been analyzed to determine which event could potentially induce the largest reduction (delta CPR) in the initial CPR. The core-wide transients which resulted in the largest delta CPR from rated conditions (104% power/100% flow) are the load rejection without bypass (LRWOB) and the feedwater controller failure (FWCF). These resulted in delta CPRs of 0.24 and 0.23, respectively. The most limiting local transient, the rod withdrawal error (RWE), was analyzed to support a rod block monitor (RBM) setting of 108% and resulted in a delta CPR of 0.26, requiring a MCPR operating limit of 1.32. This was the most limiting event for S2C3 at rated power and flow conditions. At less than rated power, the FWCF event is limiting and a curve of MCPR versus power 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 control failure is limiting and MCPR operating limits for manual flow control reduce flow operation for Cycle 3 based on the analysis of this event are provided as a Technical Specification figure of MCPR versus core flow. These calculations were performed with approved methodology 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 the MCPR operating limits with either of these systems inoperable. This resulted in increased MCPR limits which are also proposed for S2C3. These calculations follow standard procedures and operation within the proposed MCPR operating limits with either the main turbine bypass inoperable or the end-of-cycle RPT inoperable is acceptable for S2C3.

Compliance with overpressurization criteria was demonstrated by analysis of main steam isolation valve (MSIV) closure with MSIV position switch scram failure. Six safety relief valves were assumed out of service. Maximum pressure was 104% of vessel design pressure, well within the 110% criterion. The calculation was done with approved methodology and the results are acceptable.

5.2 Postulated Accidents

The GE loss of coolant accident (LOCA) analysis and maximum average planar linear heat generation rate (MAPLHGR) limits for the GE 8X8 fuel remain applicable for Cycle 3 although an additional exposure point at 40,675 MWD/MTU is added to the GE Type III MAPLHGR limit curve. The staff has previously approved this new curve for S1C3 operation and finds it acceptable for GE Type III fuel in S2C3 as well. The licensee has also presented MAPLHGR limits for the ANF 9X9 fuel based on the analysis results provided in XN-NF-86-65. The LOCA analyses have covered an acceptable range of conditions, have been performed with approved methodology, and the results meet the limits specified in 10 CFR 50.46. Therefore, the staff finds the proposed MAPLHGR limits for S2C3 acceptable.

The control rod drop accident (CRDA) was analyzed with approved ANF methodology. The maximum fuel rod enthalpy was 205 cal/gm, which is well below the design limit of 280 cal/gm, and less than 250 fuel rods exceeded 170 cal/gm, which is less than the 770 rods assumed in the SSES FSAR analysis. To ensure compliance with the CRDA analysis assumptions, control rod sequencing below 20% core thermal power must comply with GE's banked position withdrawal sequencing constraints. The staff finds the analysis and results of the CRDA for Cycle 3 acceptable.

6.0 TECHNICAL SPECIFICATION CHANGES

The following Technical Specification (TS) changes have been proposed for SSES for operation during reload Cycle 3:

(1) TS 3/4.2.1, Average Planar Linear Heat Generation Rate

The allowed exposure for GE 2.33 w/o enriched fuel has been increased to 40,675 MWD/MTU. In addition, editorial changes to correct misarranged wording and the vendor reference are made.

The change to the GE limit is based on an approved GE LOCA analysis and is acceptable as discussed in Section 4.3 of this SER. The editorial changes include the replacement of references to "Exxon" with "ANF" to reflect the corporate name change. These editorial changes are administrative only with no safety significance and are, therefore, acceptable.

(2) TS 3/4.2.2, APRM Setpoints

An editorial change corrects the vendor reference from "Exxon" to "ANF". This change is administrative only with no safety significance and is acceptable.

(3) TS 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 NRC criteria. Therefore, the revised MCPRs are acceptable for Cycle 3 as discussed in Section 5.0 of this SER.

(4) TS 3/4.2.4, Linear Heat Generation Rate

An editorial change corrects the vendor reference from "Exxon" to "ANF."

This change is administrative only with no safety significance and is acceptable.

(5) TS 3/4.3.6, Control Rod Block Instrumentation

Footnote "##" to trip function 2a has been added to refer to TS 3.4.1.1.2.a for single loop operation requirements.

This change is editorial in nature and since single loop operation has been approved for S2C3 in the staff's safety evaluation for single loop operation, it is acceptable.

(6) TS 3/4.4.1, Recirculation System

Changes have been made to the single loop and two loop operation requirements.

These changes have been reviewed and evaluated under a separate licensing action.

Based on the above considerations the staff 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 and are acceptable.

3.0 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. 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 (53 FR 2322) on January 27, 1988 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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