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

November 3, 1989

Docket No. 50-388

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 TO SUPPORT CYCLE 4 OPERATION (TAC NO. 73588)

RE: SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station (SSES), Unit 2. This amendment is in response to your letter dated June 16, 1989 as clarified by your letter dated October 6, 1989.

This amendment changes the SSES, Unit 2 Technical Specifications in support of the fuel reload for Cycle 4 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,

Mohan C. Thadani

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

Enclosures:
1. Amendment No. 58 to License No. NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

CP 1

Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station
Units 1 & 2

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November 1989

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Pennsylvania Power and Light Company
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/s/
Mohan C. Thadani, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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2. Safety Evaluation

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See next page

[73588 LETTER]

PDI-2/LA	PDI-2/PM	PDI-2/D	OGC
MO'Brien	MThadani:tr	for WButler	
11/3/89	10/26/89	10/3/89	10/31/89

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. 58
License No. NPF-22

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated June 16, 1989 as clarified October 6, 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-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. 58 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 3, 1989

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	MThadani		WButler
11/3/89	10/26/89	10/3/89	10/3/89

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

FOR THE NUCLEAR REGULATORY COMMISSION

for Mohan C Thadani

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 3, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 58

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
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 2-7
-	3/4 2-8
3/4 2-7	3/4 2-9
3/4 2-8	3/4 2-10
3/4 2-9	-
3/4 2-10	-
3/4 2-10a	-
3/4 2-10b	-
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
3/4 4-1d	3/4 4-1d
3/4 4-1e	3/4 4-1e*

ATTACHMENT TO LICENSE AMENDMENT NO. 58

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

REMOVE	INSERT
3/4 4-1f	3/4 4-1f*
3/4 4-1g	-
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
-	-
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2*
5-5	5-5*
5-6	5-6

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ADMINISTRATIVE CONTROLS

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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 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 9 x 9 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For this design, the coolant minimum flow and maximum flow area is such that the mass flux is always greater than 0.25×10^6 lbs/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lbs/hr-ft^2 is 3.35 Mwt or greater. At 25% thermal power a bundle power of 3.35 Mwt corresponds to a bundle radial peaking factor of 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 all fuel shall not exceed the limit shown in Figure 3.2.1-1.

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

ACTION:

With an APLHGR exceeding the limit of Figure 3.2.1-1, initiate corrective action within 15 minutes and restore APLHGR 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

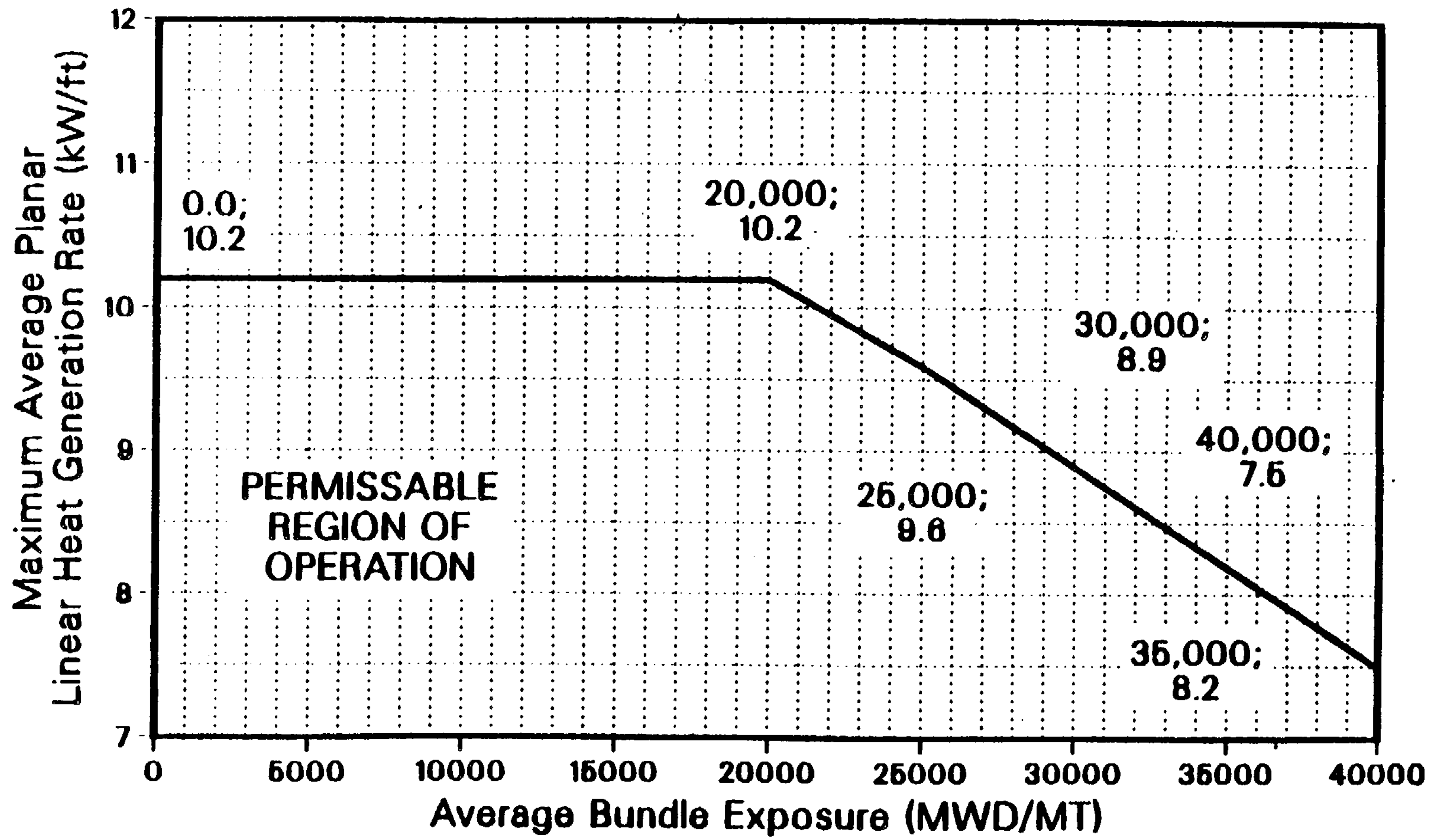
4.2.1 All APLHGRs shall be verified to be equal to or less than the limit determined from Figure 3.2.1-1:

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
ANF 9X9 FUEL
FIGURE 3.2.1-1

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 FRACTION OF LIMITING POWER DENSITY (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

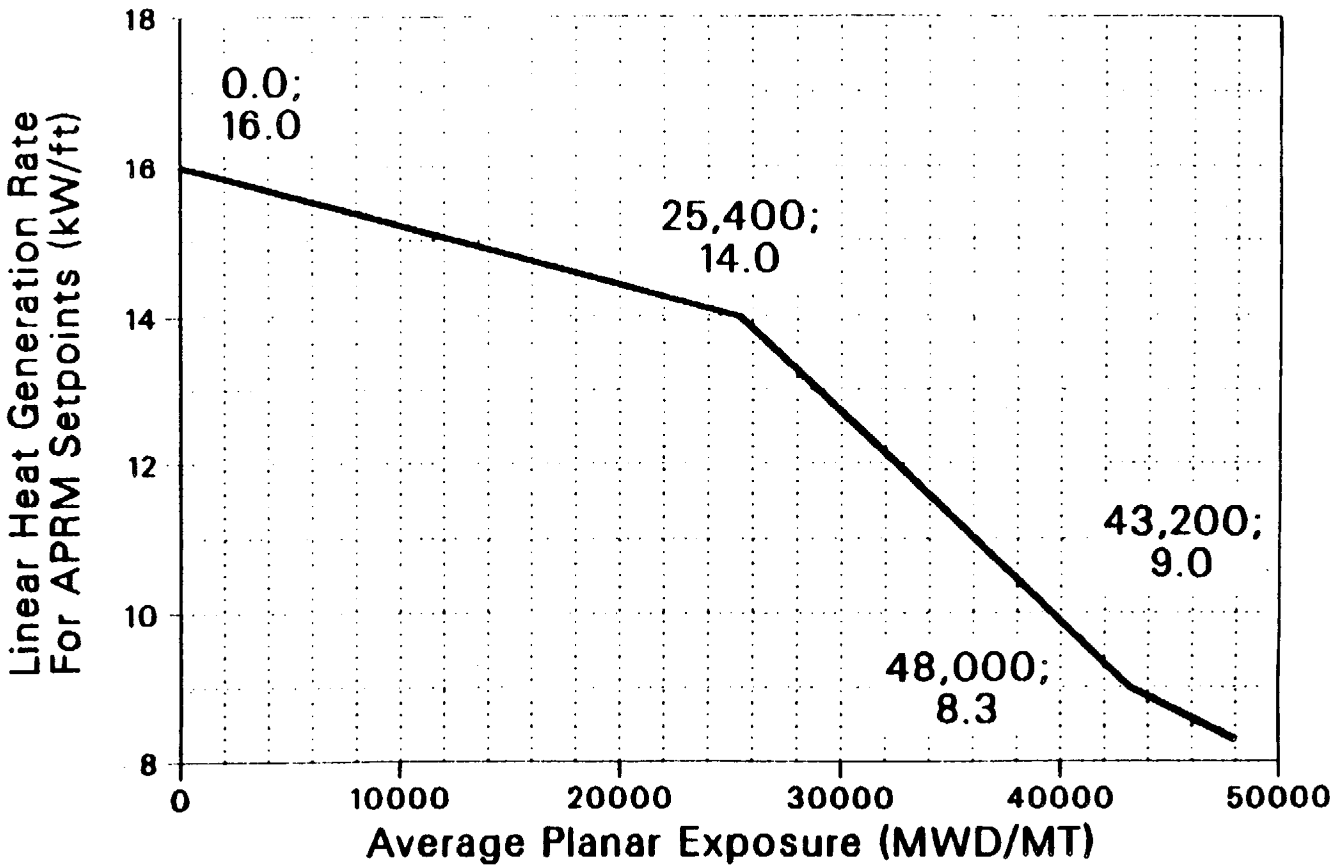
4.2.2 (Continued)

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.



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

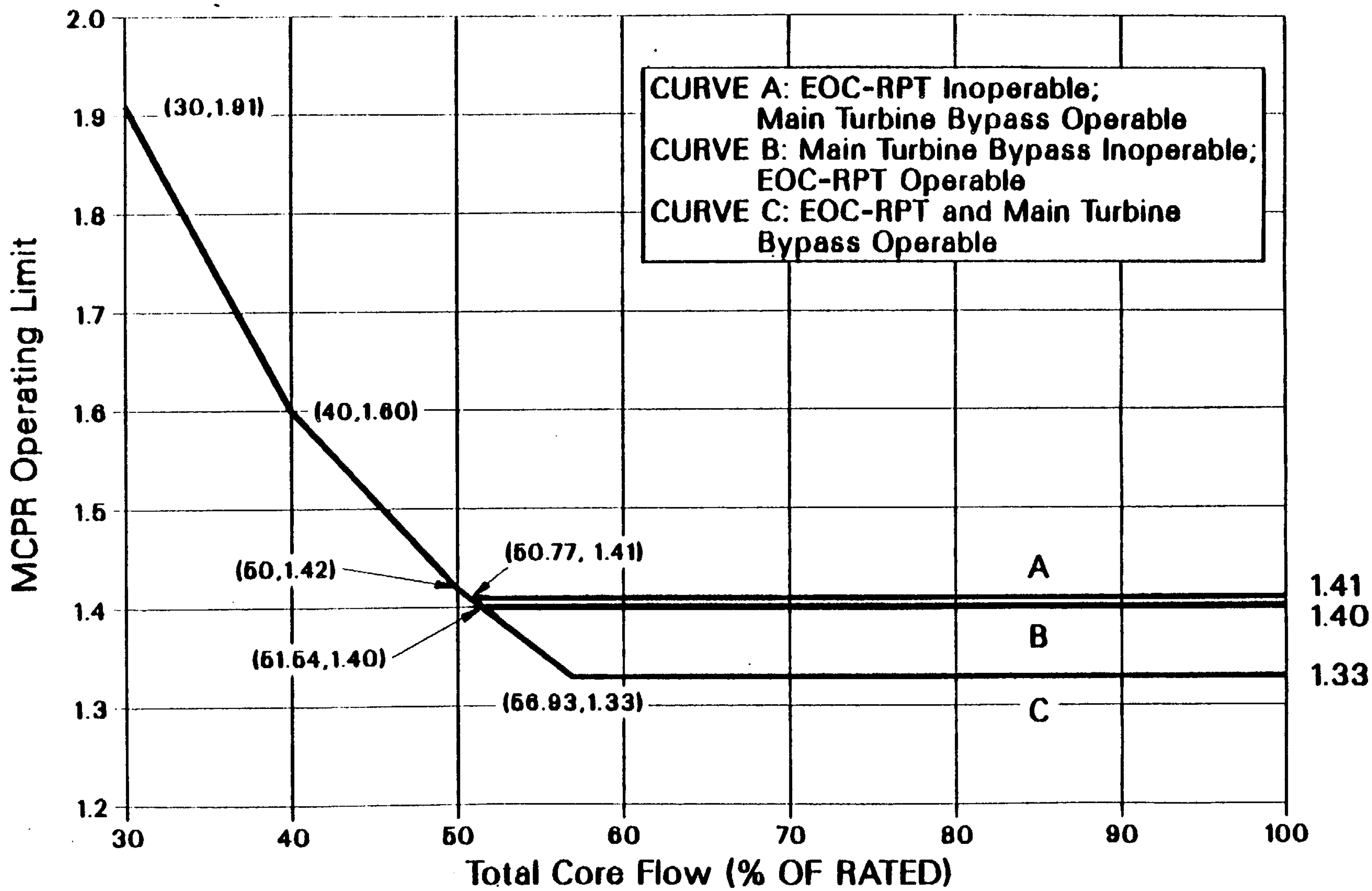
4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit determined from Figure 3.2.3-1 and Figure 3.2.3-2:

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

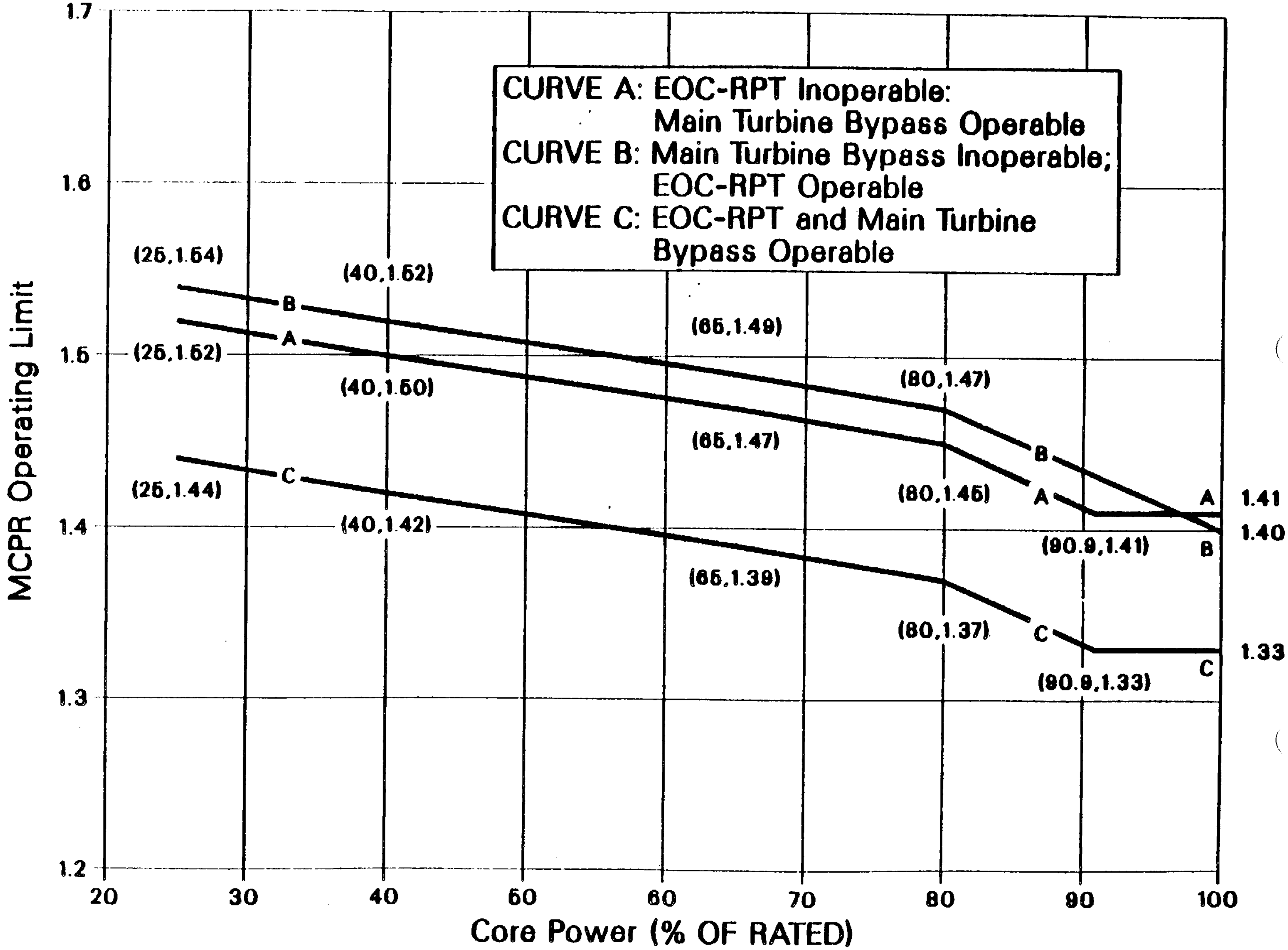
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

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

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

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

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

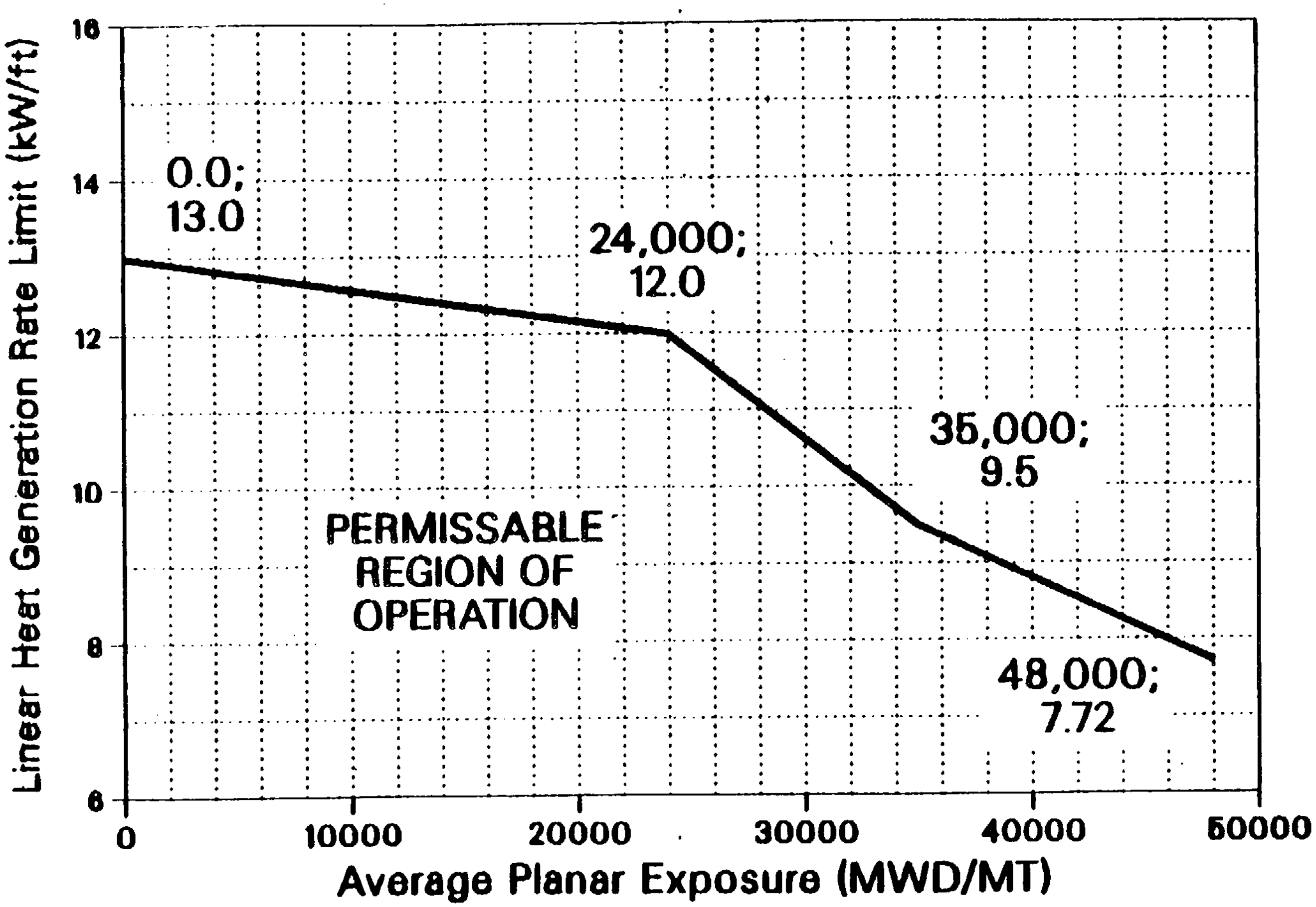
4.2.4 LHGRs shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.



LINEAR HEAT GENERATION RATE (LHGR) LIMIT
VERSUS AVERAGE PLANAR EXPOSURE
ANF 9X9 FUEL
FIGURE 3.2.4-1

3/4.4 REACTOR COOLANT SYSTEM

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 the reactor shall be 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 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. 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 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 or 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

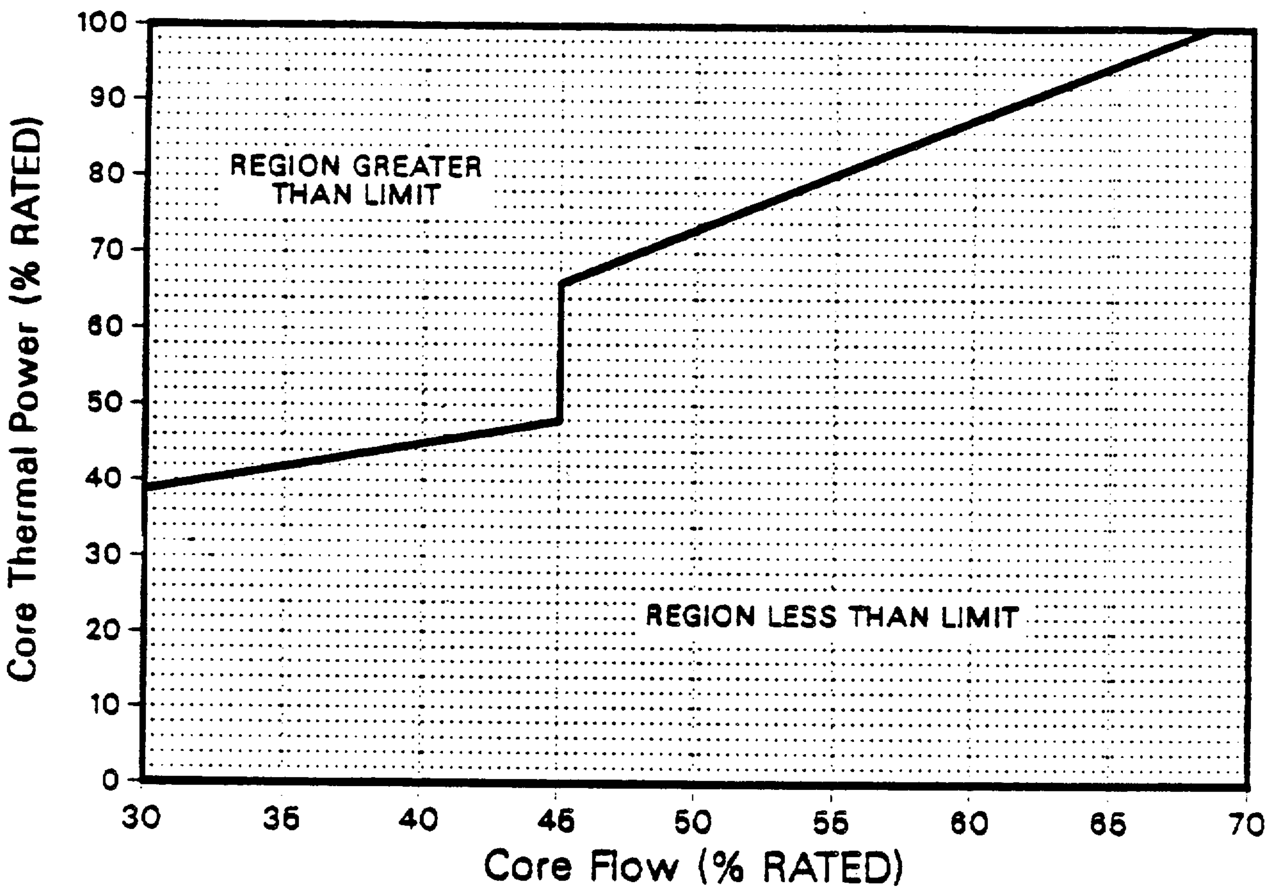

Figure 3.4.1.1.1-1
THERMAL POWER/CORE FLOW LIMITATIONS

RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed ≤ 80% of the rated pump speed and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.
2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

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

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

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

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the largest of the following values:
   a. the MCPR determined from Figure 3.2.3-1 plus 0.01, and
   b. the MCPR determined from Figure 3.2.3-2 plus 0.01.
5. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

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

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 Figure 3.4.1.1.1-1.

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 references 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 3.4.1.1.1-1, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours 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. 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. Returning the rector to a THERMAL POWER/core flow condition less than or equal to the limit specified in Figure 3.4.1.1.1-1 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 3.4.1.1.1-1, 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 $\leq$ 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.

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The Technical 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 Figure 3.2.1-1.

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

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analyses. In addition, the APRM setpoints must be adjusted to ensure that $\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 AOO's.

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 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, Rod Withdrawal Error, or Load Reject Without Main Turbine Bypass operable initiated from a reduced power condition.

Cycle specific analyses are performed for the most limiting local 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.

MINIMUM CRITICAL POWER RATIO (Continued)

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

3/4.2.4 LINEAR HEAT GENERATION RATE

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

3/4.4.1 RECIRCULATION SYSTEM

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

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

The MINIMUM CRITICAL POWER RATIO (MCPR) limits for single loop operation assure that the Safety Limit MCPR is not exceeded for any Anticipated Operational Occurrence (AOO).

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

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

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.

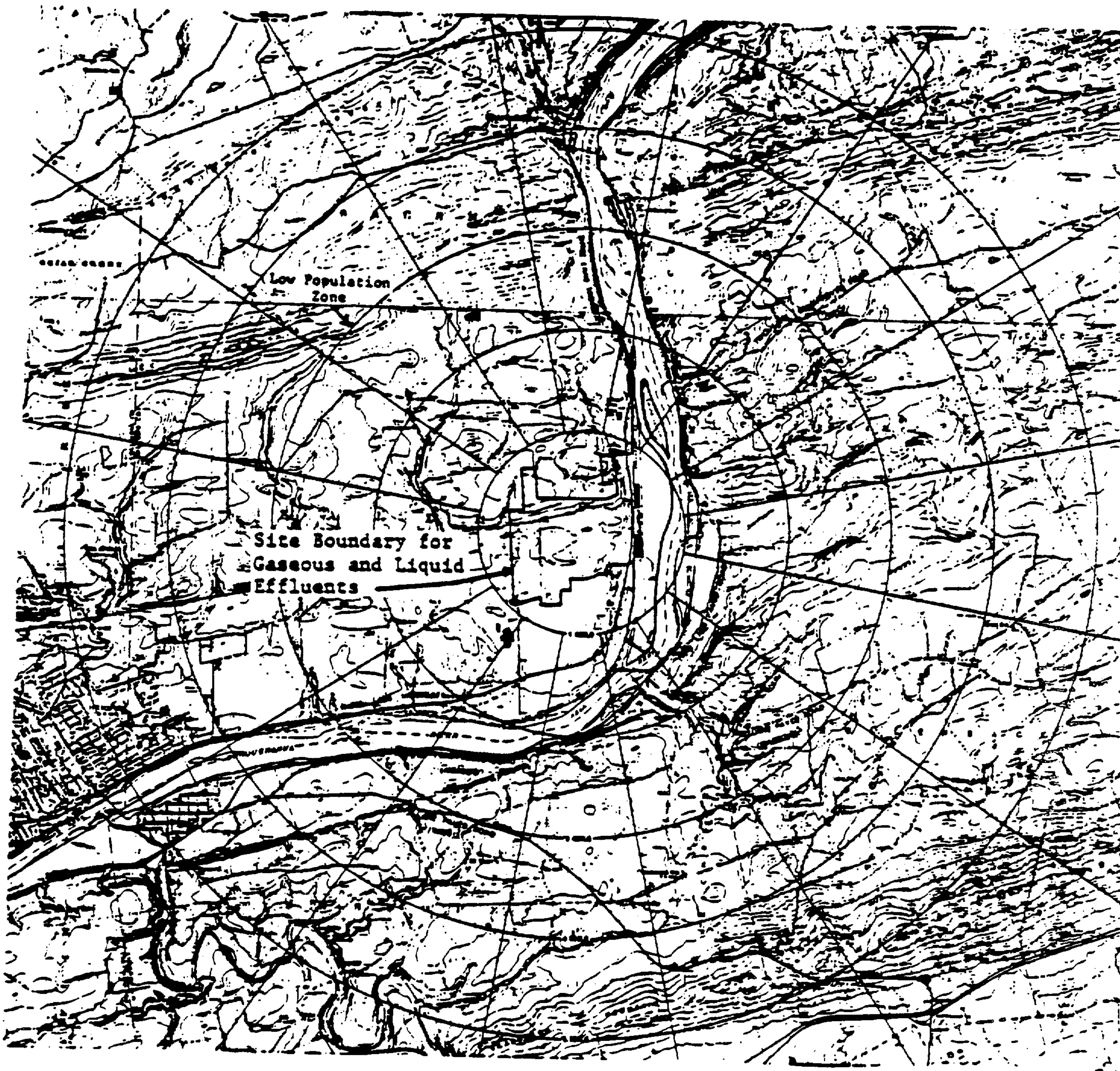

FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED AREAS
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

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

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

b. For a pressure of:

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

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

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal T_{ave} of 528°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 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

In a letter dated June 16, 1989, Pennsylvania Power and Light Company (PPLC) proposed Technical Specification (TS) changes to support operation of Susquehanna Steam Electric Station (SSES), Unit 2 for Cycle 4 (S2C4) with a reload using Advanced Nuclear Fuel (ANF) manufactured fuel assemblies and ANF analyses and methodologies. Enclosed were the proposed Technical Specification (TS) changes and reports discussing the reload and analyses done to support and justify Cycle 4 operation, including single (recirculation) loop operation (SLO). There was an additional submittal dated October 6, 1989 providing clarification of the recirculation pump seizure event for SLO in response to NRC staff questions. The submittal did not affect the staff's proposed no significant hazards determination.

The Cycle 4 reload will replace 208 General Electric (GE) fuel assemblies, used in Cycle 3, with 204 ANF-3 and 4 XN-1 assemblies from ANF. For the first time there will be no GE fuel in SSES, Unit 2. All fuel will be ANF 9x9 assemblies, with 556 ANF XN-2 or XN-1 assemblies remaining from the previous cycle. The reload for Cycle 4 is generally a normal reload with no unusual core features or characteristics other than the complete ANF 9x9 loading. TS changes are not extensive and are primarily related to Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) for the ANF fuel and Minimum Critical Power Ratio (MCPR) limits for Cycle 4 core operation. Many of the TS changes are of an administrative nature, removing references to GE fuel, limits and methodology.

There are several proposed modifications to the current TS for SLO and two loop operation (TLO). A few of these changes are related to thermal hydraulic stability (THS) Specifications. However, the THS Specifications have been completely revised in a separate submittal (Ref. 20) and these have been separately reviewed (Ref. 21). This will be discussed later in this review.

2.0 EVALUATION

2.1 Fuel Design

The S2C4 core reload will include 204 ANF 9x9 fuel bundles with the designation ANF-3. These reload bundles contain 79 fuel rods and 2 water rods. The 204 fuel bundles will have a bundle average enrichment of 3.33 or 3.17 weight percent uranium-235. The fuel design and safety analysis are described in the Susquehanna 2 specific report PL-NF-89-003 (Ref. 2) and the generic ANF 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). The reload fuel for S2C4 is similar in its mechanical design features to the ANF 9x9 fuel approved and used in Cycle 3 and in SSES, Unit 1, Cycle 5.

Table 2.1 of Reference 5 gives the pertinent design data for ANF 9x9 fuel. Neutronic values specific to the S2C4 reload are given in Table 4.1 of ANF-89-058 (Ref. 4). The burnable poison rods contain 4.0 or 5.0 weight percent gadolinia blended with 3.27 or 3.10 weight percent uranium-235. The ANF-3 fuel bundles are designed to fit into channel boxes corresponding to the design of those used for the GE fuel. The analyses for S2C4 support fuel bundle discharge exposures of 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 S2C4 reload to be acceptable.

Calculation of the fuel rod internal pressure has been done in accordance with acceptance criteria cited by ANF in Reference 5. The evaluation was performed with the RODEX2A computer code which has been reviewed and approved by the staff (Ref. 8). The staff has concluded that the acceptance criteria for rod internal pressure can be fully met throughout the entire expected irradiation life of the 9x9 fuel.

A figure of LHGR limit versus planar exposure (MWd/MTU) for the ANF 9x9 fuel is incorporated into the SSES, Unit 2 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 Reference 5. Based on the results of the generic review, the staff finds the LHGR limits for the 9x9 fuel to be acceptable.

The licensee has discussed the mechanical response of the ANF 9x9 fuel assembly design during LOCA-seismic events in Appendix B of Reference 4. The discussion includes a comparison of the physical and structural properties of the ANF 9x9 fuel and the GE 8x8 fuel. The staff has reviewed this information in connection with a previous review (see Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 31 to Facility Operating License No. NPF-22 dated October 3, 1986). The staff has confirmed that the physical and structural characteristics of the ANF and GE fuel assemblies are sufficiently similar so that the mechanical response to design LOCA-seismic events is essentially the same. Based on the considerations discussed above, we conclude that the original analysis is applicable to SSES, Unit 2 and the analysis indicating that the design limits are not exceeded is acceptable.

2.2 Nuclear Design

The ANF nuclear design methodology for S2C4 is presented in XN-NF-80-19(A), Volume 1 and Volume 1 Supplements 1 and 2 (Ref. 9), which were reviewed and approved by the staff for generic application to BWR core reloads. The S2C4 licensing analysis employed a new cross section-void history correlation different from that used in Reference 9. 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. 10) 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 1.29 percent delta-k/k, and the R factor is 0.01 percent. 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 1.0998, the ANF calculations show that the staff's acceptance criterion is met for the new fuel storage vault under dry and flooded conditions. To preclude criticality at optimum moderation conditions (between dry and flooded), watertight covers and appropriate procedures are used. These are acceptable.

ANF also performed analyses for 9x9 fuel stored in the spent fuel pool. A maximum enriched zone of less than 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-3 9x9 fuel has a zone average enrichment of 3.44 weight percent uranium-235 and a maximum k-infinity of 1.2020 at peak reactivity exposure, the staff's acceptance criterion for spent fuel storage is met for the ANF-3 9x9 fuel.

SSES 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 SSES Unit 1 and Unit 2 and has provided acceptable monitoring and predictive results.

2.3 Thermal-Hydraulic Design

The review of the thermal-hydraulic aspects of ANF reloads normally consists of: (a) the compatibility of the ANF reload and prior fuel assemblies, (b) the fuel cladding integrity safety limit, (c) the bypass flow characteristics, and (d) thermal hydraulic stability. However, S2C4 consists of all ANF 9x9 fuel, and there is no compatibility concern, and the stability has been reviewed elsewhere (as discussed later).

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. 11), which has been approved by the staff (Ref. 12). The XN-3 correlation used to develop the MCPR safety limit has been approved for the ANF 9x9 fuel (Ref. 13). The methodology of XN-NF-524, Revision 1 was applied generically for S2C4 and the staff has verified through its review of the S2C4 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.

The core bypass flow fraction has been calculated to be 10.0 percent of total core flow using the approved methodology described in XN-NF-524 (P)(A), Revision 1 (Ref. 11). This is used in the MCPR safety limit calculations and as input to the S2C4 transient analyses and is acceptable.

The S2C4 submittal addressed thermal hydraulic stability (THS), and several of the proposed (minor administrative type) changes to TS 3/4.4.1.1.1 and .2 were THS related specifications. However, a subsequent submittal for S2C4 (Ref. 20) discussed more extensive proposed THS TS changes (primarily involving the above TS). These were intended to provide compatibility with the procedures installed to comply with the NRC requests related to THS concerns as presented in NRC Bulletin 88-07 and Supplement 1 to that bulletin. That submittal has been reviewed separately (Ref. 21), and THS TS changes approved. With those changes the S2C4 THS operations and TS are currently considered to be acceptable and the discussion in the reload submittal (which is no longer directly relevant) will not be considered further in this review. However, this acceptance is based on preliminary assessment of the stability characteristics of ANF 9x9 fuel based on neutron flux noise data obtained during SSES, Unit 2 Cycles 2 and 3. Those tests generally confirmed stability characteristics with 9x9 fuel assembly loadings comprising up to about 70 percent of the core. Since for Cycle 4 the core will contain all 9x9 fuel for the first time, it will be necessary to complete the stability examination with similar measurements during Cycle 4. (This was indicated during the 1988 staff review of SLO for SSES, Unit 2.) The measurements should be made during the closest approach to Region I of TS Figure 3.4.1.1.1-1, during initial startup and when reasonably possible at burnup intervals during the cycle, and the data presented to the NRC.

2.4 Transient and Accident Analyses

Various operational transients could reduce MCPR below the safety limit. The most limiting transients have been analyzed to determine which event could potentially result in the largest reduction in the initial 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 the ANF 9x9 fuel. 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 0.26. The LRWOB and the RWE events were the most limiting events for S2C4 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 S2C4 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. 14) and the resulting Technical Specification limiting curves are acceptable.

It was assumed for the above analyses that (1) the turbine bypass system and (2) 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 (from 1.33 to 1.40 and 1.41 for (1) and (2) respectively) MCPR limits which are also proposed for S2C4. 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 S2C4.

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, which is more conservative than the 3.0 seconds used in previous analyses. 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.

The LOCA analyses for the SSES, Units 1 and 2 (Ref. 15) were performed for a full core of ANF 9x9 fuel and is applicable for the S2C4 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 MAPLHGP values for the ANF fuel remain acceptable.

The control rod drop accident (CRDA) was analyzed with approved ANF methodology (Ref. 9). The maximum fuel rod enthalpy was 249 cal/gm, which is well below the design limit of 280 cal/gm, and less than 650 fuel rods exceed 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 percent core thermal power must comply with GE's banked position withdrawal sequencing constraints (Ref. 16). The staff concludes that the analysis and results for the S2C4 CRDA are acceptable.

2.5 Single Loop Operation (SLO)

Current Technical Specifications for SSES Unit 2 permit plant operation with a single recirculation loop out-of-service for an extended period of time. Analyses for S2C4 (Ref. 4) show that the MCPR Safety Limit must be increased by 0.01 because of the increased measurement uncertainties, and this is already included in the SLO TS. Previous analyses reported by the licensee (Refs. 17 and 18) have shown that events, other than (possibly) recirculation pump seizure, which could be affected by SLO are non-limiting when analyzed under SLO conditions.

The pump seizure event is more severe under SLO than under two loop operation, assuming pump seizure of the operating loop. The S2C4 submittal indicated that, unlike previous SSES submittals in which PPLC had chosen to consider this event to be an "anticipated operational occurrence," it was decided, in keeping with the NRC Standard Review Plan requirements, to treat the event as an "accident." There would therefore be no requirement not to exceed the safety limit MCPR for the event. There is, however, a Standard Review Plan requirement to determine the amount of "fuel failure" calculated for the event (number of fuel pins exceeding the safety limit MCPR) and to limit any resulting calculated radiation dose to a "small fraction of 10 CFR 100 guidelines." The staff review indicated that the information provided in the original submittal was unclear and following a discussion between PPLC and the staff there was a second submittal (Ref. 19). This described a more complete analysis of the event and consequences. The result for S2C4 indicates that this event will easily meet the "small fraction of 10 CFR 100 guidelines," and that in fact it will not exceed the safety limit MCPR since it has a delta MCPR less than FWCF which sets the MCPR limit at less than rated power (in the regime applicable to SLO). This analysis and result is acceptable.

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.

SLO for S2C4 must maintain the 80 percent recirculation pump speed restriction because of the previous GE vessel internal vibration analysis, as discussed in Reference 17.

2.6 Technical Specification Changes

The following Technical Specification (TS) changes have been proposed for operation of S2C4:

(1) Index

There are administrative changes relating to removal of GE fuel, page numbers and Figure reductions. They are relevant to approved changes and are acceptable.

(2) Basis 2.0 Introduction

Editorial changes are made to this Basis. The reference to GE fuel is deleted. S2C4 will not contain GE fuel. Revision 1 is appended after report XN-NF-524(A). These changes are editorial changes and are, therefore, acceptable.

(3) Basis 2.1.1 Thermal Power, Low Pressure or Low Flow

The reference to GE and 8x8 is deleted since S2C4 will not contain these types of fuel. This is an editorial change 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, including Figures for GE fuel, are deleted because S2C4 will not contain GE fuel. Also the footnote reference to SLO is removed since it was only applicable to GE fuel. These are editorial changes and are, therefore, acceptable.

(6) Specification 3/4.2.2 APRM Setpoints

The reference to FLPO for GE fuel is deleted because S2C4 will not contain any GE fuel. There are also page number changes. These changes are editorial and are, therefore, acceptable.

(7) Specification 3/4.2.3 Minimum Critical Power Ratio

Operating limit MCPRs displayed in Figures 3.2.3-1 and -2 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 S2C4 as discussed in this SER.

(8) <u>Specifications 3/4.2.4.1 and .2 Linear Heat Generation Rate</u>

The limit and reference to GE fuel are being deleted by removing 3/4.2.4.1 and deleting vendor references in 3/4.2.4.2, which becomes 3/4.2.4 because S2C4 will not contain any GE fuel. These changes are editorial changes and are, therefore, acceptable.

(9) <u>Specification 3.4.1.1.1 Two Loop Operation</u>

The references to 55 million lbs/hr core flow have been removed. Reference is now only to the curve in Figure 3.4.1.1.1-1 which provides the correct limit. The curve has been extended to cover operation up to full power. The 55 million lb/hr limit was only a quick, approximate reference for the operator. The change is editorial and is acceptable. This will be changed again by the THS TS review discussed in Section 2.3 of this review.

(10) <u>Specification 3.4.1.1.2 Single Loop Operation</u>

There are several changes to this TS.

(a) 3.4.1.1.2.a.2, .4 and .6 provide trip setpoint modification for operation during SLO for APRM flow-biased scram (Table 2.2.1-1), for APRM (TS 3.2.2) and for RBM/APRM Rod Block (Table 3.3.6-2). This is required to account for the change from 7 to 8.5 in drive flow for a given core flow discussed in Section 2.5 of this review. This change is acceptable.

(b) 3.4.1.1.2.a.3 provides a MAPLHGR reduction factor for GE fuel during SLO. It is deleted since there is no GE fuel in S2C4. This change is acceptable.

(c) 3.4.1.1.2.a.5.a provides an operating limit MCPR based on the SLO pump seizure event analysis. It has been deleted since, as discussed in Section 2.5 of this review, this event is not limiting for S2C4. This change is acceptable.

(d) Figure 3.4.1.1.2-1 has been deleted and all references to that figure are changed to Figure 3.4.1.1.1-1. This change is acceptable, however, it is irrelevant since, as discussed in Section 2.3, new TS have been separately reviewed and approved for THS, and these TS will replace the specifications referring to this figure.

(e) Action sections c and e have administrative changes indicating both power and flow changes may be used to provide stability. This is acceptable, but as above, irrelevant since the new approved THS TS will change this section completely.

(11) Bases 3/4.2.1, .2 and .3 and References.

References to GE fuel are deleted, which is acceptable since there is no GE fuel in S2C4 and minor editorial changes have been made which do not affect the content of the Bases.

(12) Basis 3/4.4.1 Recirculation System

The Basis is changed to reflect that the MAPLHGR multiplier for ANF fuel for SLO is 1.0 and a multiplier is no longer needed for GE fuel. 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 PPLC determined value of 8.5 percent. These changes are acceptable because the analysis for SLO for S2C4 was performed using approved methodologies which gave acceptable results.

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 (54 FR 31110) on July 26, 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.

5.0 REFERENCES

1. Letter (PLA-3209) from H.W. Keiser (PPLC) to W.R. Butler (NRC), "Proposed Amendment 74 to License No. NPF-22: Unit 2 Cycle 4 Reload," dated June 16, 1989.

2. PL-NF-89-003, "Susquehanna SES Unit 2-Cycle 4: Reload Summary Report," dated May 1989.

3. ANF-89-057, "Susquehanna Unit 2 Cycle 4: Plant Transient Analysis," dated May 1989.

4. ANF-89-058, "Susquehanna Unit 2 Cycle 4 Reload Analysis: Design and Safety Analyses," dated May 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. 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.

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

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

12. 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.

13. 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.

14. 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.

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

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

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

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

19. Draft Telefax Note, October 2, 1989, from PPLC to M. Thadani, NRC, "Susquehanna Unit 2 Single Loop Pump Seizure Accident Analysis."

20. Letter from H. Keiser, PPLC, to W. Butler, NRC, dated June 23, 1989, "Proposed Amendment 75 to License No. NPF-22: Unit 2 Stability."

21. Memorandum to W. Butler, NRC, from M. Hodges, NRC, dated September 15, 1989, "Susquehanna 2 Technical Specification Revisions - Thermal Hydraulic Stability."

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