Docket No. 50-388

April 22, 1991

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

Dear Mr. Keiser:

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2, CYCLE 5 RELOAD (TAC NO. 79140)

The Commission has issued the enclosed Amendment No. 76 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. This amendment is in response to your application dated March 7, 1991.

This amendment would change the Technical Specifications in support of the ensuing Cycle 5 reload with ANF fuel and replacement of control rod blades.

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/
James J. Raleigh, Acting Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 76 to License No. NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	GHill(4), P1-37
NRC & Local PDR	Wanda Jones, 7103
PDI-2 Reading	JCalvo, 11F-22
SVarga	LKopp
EGreenman	ACRS(10)
WButler	GPA/PA
MO'Brien(2)	OC/LFMB
JRaleigh	RBlough, RGN-I
JStone	JWhite, RGN-I
OGC	
DHagan, MS-3206	

*See Previous Concurrence

OFC	:PDI-2/LA*	:PDI-2/PM*	:PDI-2/D JC for	:OGC*	:
NAME	:MO'Brien	:JRaleigh:rb	:WButler	:MYoung	:
DATE	:03/21/91	:03/21/91	04/22/91	:03/28/91	:

OFFICIAL RECORD COPY
Document Name: TAC 79140

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

April 22, 1991

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: SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2, CYCLE 5 RELOAD (TAC NO. 79140)

The Commission has issued the enclosed Amendment No. 76 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. This amendment is in response to your application dated March 7, 1991.

This amendment would change the Technical Specifications in support of the ensuing Cycle 5 reload with ANF fuel and replacement of control rod blades.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,

James J. Raleigh

James J. Raleigh, Acting Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 76 to
   License No. NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

Jay Silberg, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street N.W.
Washington, D.C. 20037

Bryan A. Snapp, Esq.
Assistant Corporate Counsel
Pennsylvania Power & Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Mr. J. M. Kenny
Licensing Group Supervisor
Pennsylvania Power & Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Mr. Scott Barber
Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P.O. Box 35
Berwick, Pennsylvania 18603-0035

Mr. Thomas M. Gerusky, Director
Bureau of Radiation Protection
Resources
Commonwealth of Pennsylvania
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Mr. Jesse C. Tilton, III
Allegheny Elec. Cooperative, Inc.
212 Locust Street
P.O. Box 1266
Harrisburg, Pennsylvania 17108-1266

Mr. S. B. Ungerer
Joint Generation
Projects Department
Atlantic Electric
P.O. Box 1500
1199 Black Horse Pike
Pleasantville, New Jersey 08232

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. Harold G. Stanley
Superintendent of Plant
Susquehanna Steam Electric Station
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Mr. Herbert D. Woodeshick
Special Office of the President
Pennsylvania Power and Light Company
1009 Fowles Avenue
Berwick, Pennsylvania 18603

Mr. Robert G. Byram
Vice President-Nuclear Operations
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

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. 76
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 March 7, 1991, 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. 76 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 liceense amendment is effective upon startup after the Unit 2 Cycle 5 Refueling Outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J Clark for

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 76

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
B 2-1	B 2-1*
B 2-2	B 2-2
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
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
3/4 4-1f	3/4 4-1f
-	-
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 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

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.

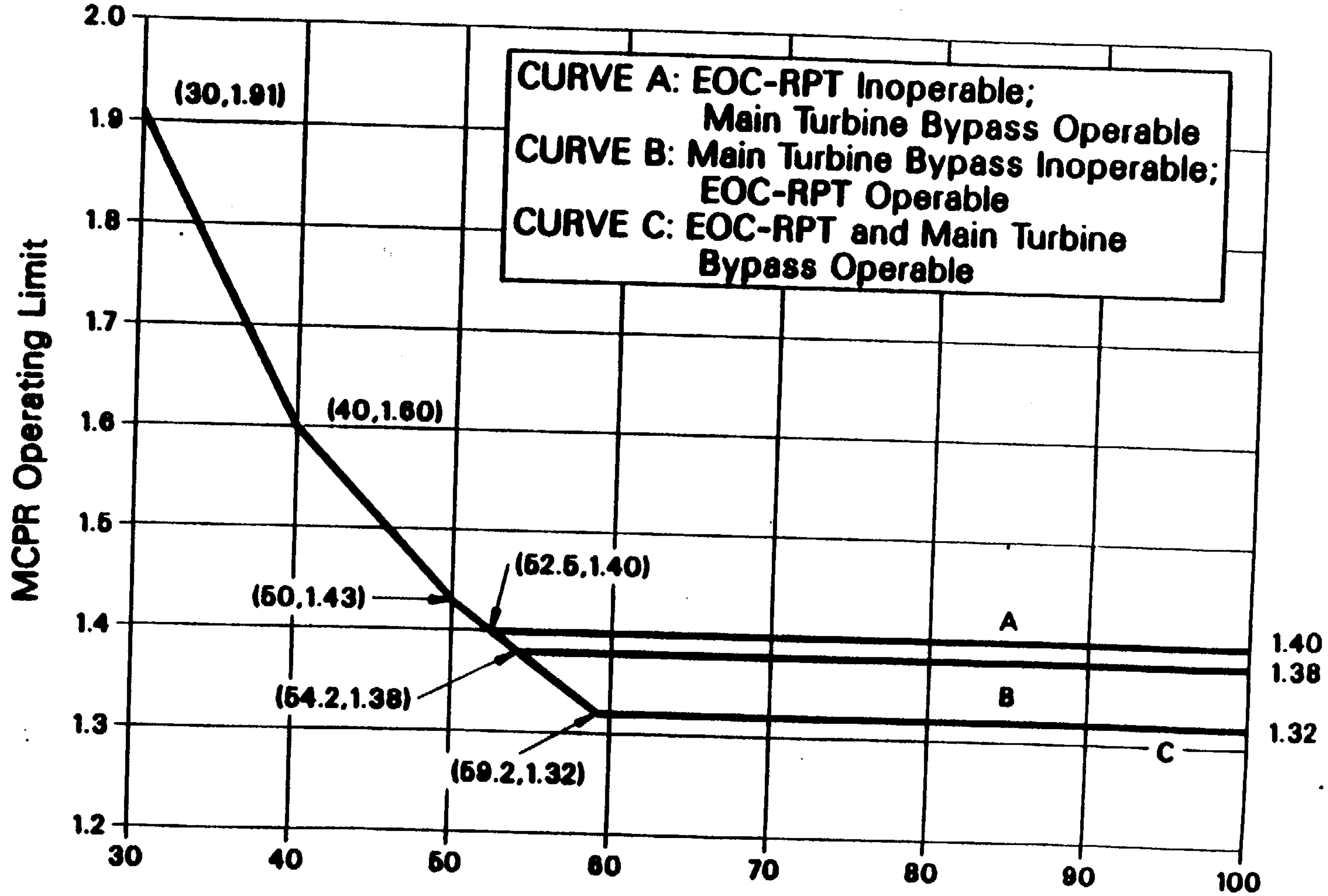
2.1.2 THERMAL POWER, High Pressure and High Flow

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).

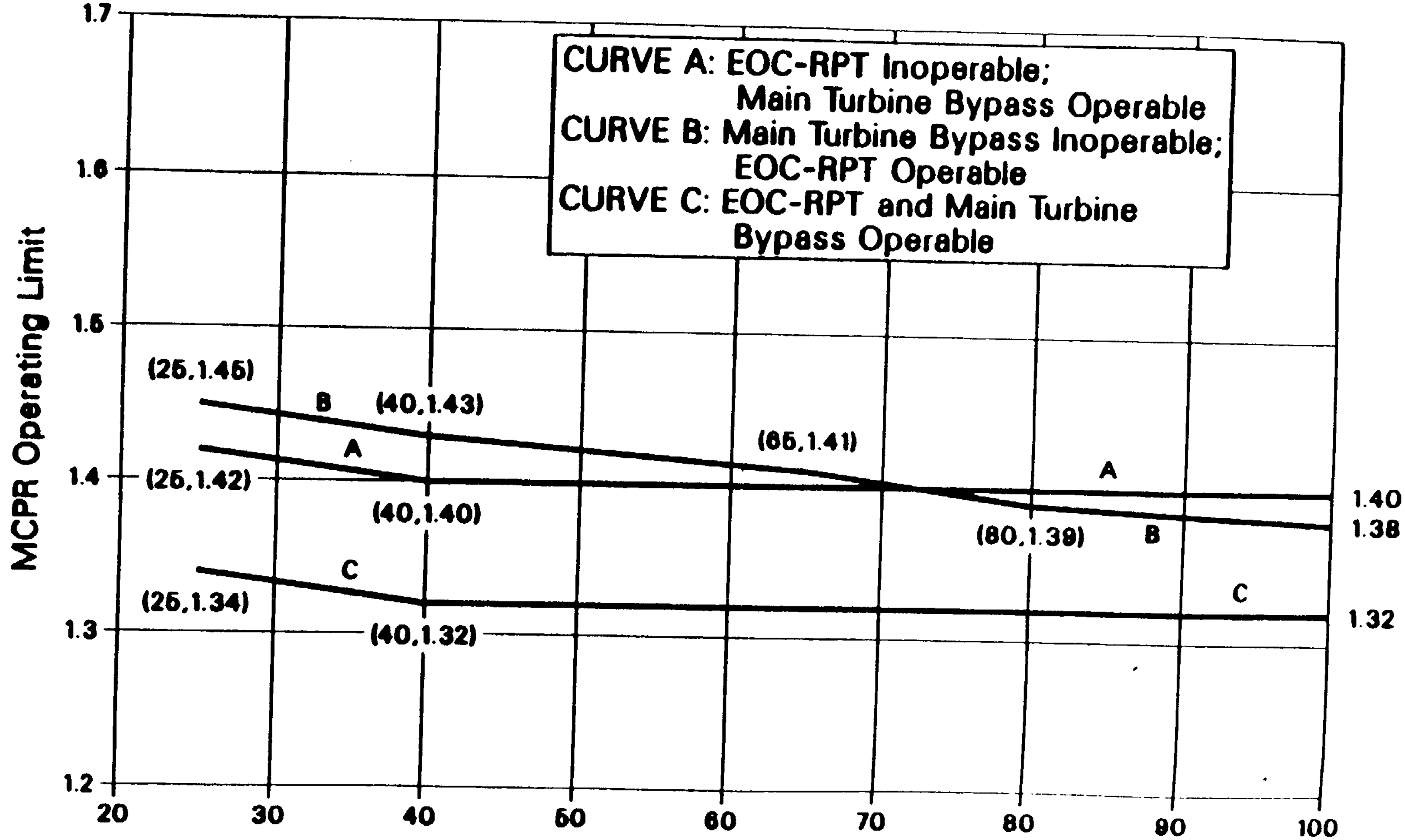
The Safety Limit MCPR assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated 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.

ANF fuel is monitored using the XN-3 Critical Power Correlation. ANF has determined that this correlation provides sufficient conservatism to preclude the need for any penalty due to channel bow. The conservatism has been evaluated by ANF to be greater than the maximum expected ΔCPR (0.02) due to channel bow in C-lattice plants using channels for only one fuel bundle lifetime. Since Susquehanna SES Unit 2 is a C-lattice plant and uses channels for only one fuel bundle lifetime, monitoring of the MCPR limit with the XN-3 Critical Power Correlation is conservative with respect to channel bow and addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."



TOTAL CORE FLOW (% OF RATED)
FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1



CORE POWER (% OF RATED)
REDUCED POWER MCPR OPERATING LIMIT
Figure 3.2.3-2

Figure 3.4.1.1.1-1
THERMAL POWER RESTRICTIONS

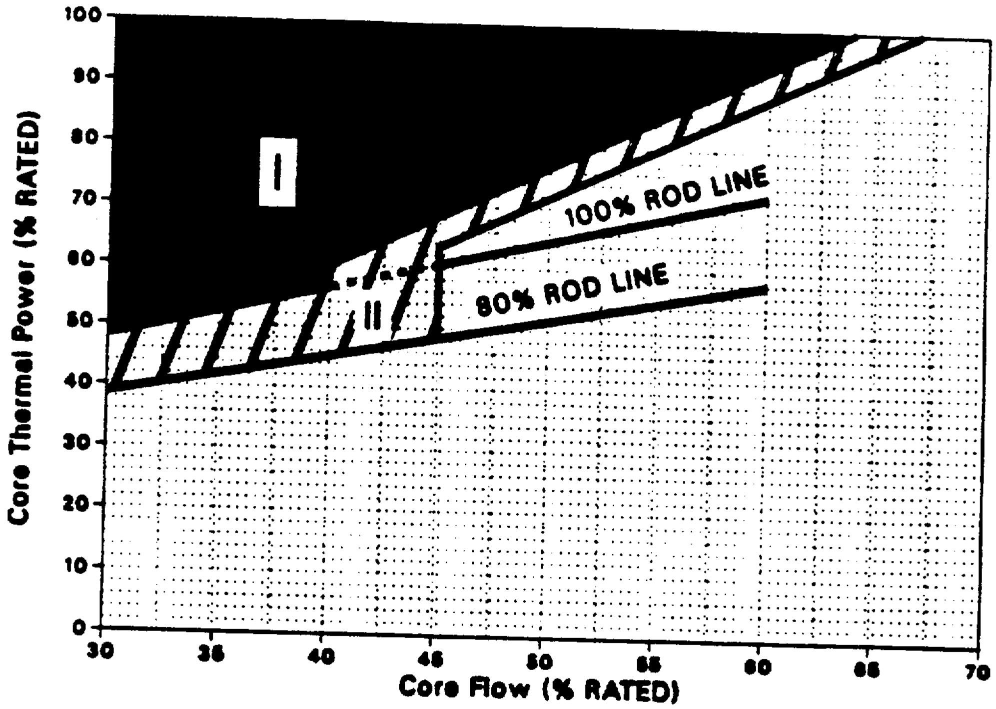


Figure 3.4.1.1.1-1
THERMAL POWER RESTRICTIONS

RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed < 80% of the rated pump speed and the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.

2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
$\leq$ 0.58W + 54%	$\leq$ 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	$\leq$ 0.66W + 36%	$\leq$ 0.66W + 39%
b. APRM-Flow Biased	$\leq$ 0.58W + 45%	$\leq$ 0.58W + 48%

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

ACTION:

a. In OPERATIONAL CONDITION 1:

1. With

   a) no reactor coolant system recirculation loops in operation, or

   b) Region I of Figure 3.4.1.1.1-1 entered, or

   c) Region II of Figure 3.4.1.1.1-1 entered and core thermal hydraulic instability occurring as evidenced by:

ACTION: (Continued)

1) Two or more APRM readings oscillating with at least one oscillating greater than or equal to 10% of RATED THERMAL POWER peak-to-peak, or

2) Two or more LPRM upscale alarms activating and deactivating with a 1 to 5 second period, or

3) Observation of a sustained LPRM oscillation of greater than 10 w/cm^2 peak-to-peak with a 1 to 5 second period, or

d) Region II of Figure 3.4.1.1.1-1 entered and less than 50% of the required LPRM upscale alarms OPERALBE,

immediately place the reactor mode switch in the shutdown position.

2. If Region II of Figure 3.4.1.1.1-1 is entered and greater than or equal to 50% of the required LPRM upscale alarms are OPERABLE, immediately exit the region by:

a) inserting a predetermined set of high worth control rods, or

b) increasing core flow by increasing the speed of the operating recirculation pump.

3. With less than 50% of the required LPRM upscale alarms OPERABLE, follow ACTION a.1.d upon entry into Region II of Figure 3.4.1.1.1-1.

b. In OPERABLE CONDITION 2 with no reactor coolant system recirculation loops in operation, return at least one reactor coolant system recirculation loop to operation, or be in HOT SHUTDOWN within the next 6 hours.

c. 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 c.1 do not apply, take the ACTION(s) required by the referenced Specification(s).

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

e. With any pump discharge valve not OPERABLE remove the associated loop from operation, close the valve and verify closed at least once per 31 days.

LIMITING CONDITION FOR OPERATION (Continued)

f. With any pump discharge bypass valve not OPERABLE close the valve and verify closed at least once per 31 days.

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 ≤ 80% of the rated pump speed.

4.4.1.1.2.2 At least 50% of the required LPRM upscale alarms shall be determined OPERABLE by performance of the following on each LPRM upscale alarm.

1) CHANNEL FUNCTIONAL TEST at least once per 92 days, and

2) CHANNEL CALIBRATION at least once per 184 days.

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 ≤ 50%**** of rated loop flow:

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

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

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

4.4.1.1.2.4 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.

4.4.1.1.2.5 The pump MG set scoop tube electrical and mechanical stops 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.

4.4.1.1.2.6 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.

SURVEILLANCE REQUIREMENTS (Continued)

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

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%.

4.4.1.1.2.7 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.

**** 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.

\+ The LPRM upscale alarms are not required to be OPERABLE to meet this specification in OPERATIONAL CONDITION 2.

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 4.4.1.1.2.6 for single loop operation requirements.

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 analyses of abnormal operational transients. For any abnormal operational transient analysis 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 settings 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 and core wide transients to determine thermal margin. Additional analyses are performed to determine the MCPR operating limit with either the Main Turbine Bypass inoperable or the EOC-RPT inoperable. Analyses to determine thermal margin with both the EOC-RPT inoperable and Main Turbine Bypass inoperable have not been performed. Therefore, operation in this condition is not permitted.

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 was made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin was demonstrated such that future MCPR evaluation below this power level is 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). In addition, the MCPR limits for single-loop operation protect against the effects of the Recirculation Pump Seizure Accident. That is, for operation in single-loop with an operating MCPR limit ≥ 1.30, the radiological consequences of a pump seizure accident from single-loop operating conditions are but a small fraction of 10CFR100 guidelines.

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

Surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive reactor vessel internals vibration. Surveillance on differential temperatures below the threshold limits 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.

Specifications have been provided to prevent, detect, and mitigate core thermal hydraulic instability events. These specifications are prescribed in accordance with NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)," dated December 30, 1988. The boundaries of the regions in Figure 3.4.1.1.1-1 are determined using ANF decay ratio calculations and supported by Susquehanna SES stability testing.

LPRM upscale alarms are required to detect reactor core thermal hydraulic instability events. The criteria for determining which LPRM upscale alarms are required is based on assignment of these alarms to designated core zones. These core zones consist of the level A, B and C alarms in 4 or 5 adjacent LPRM strings. The number and location of LPRM strings in each zone assure that with 50% or more of the associated LPRM upscale alarms OPERABLE sufficient monitoring capability is available to detect core wide and regional oscillations. Operating plant instability data is used to determine the specific LPRM strings assigned to each zone. The core zones and required LPRM upscale alarms in each zone are specified in appropriate procedures.

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.

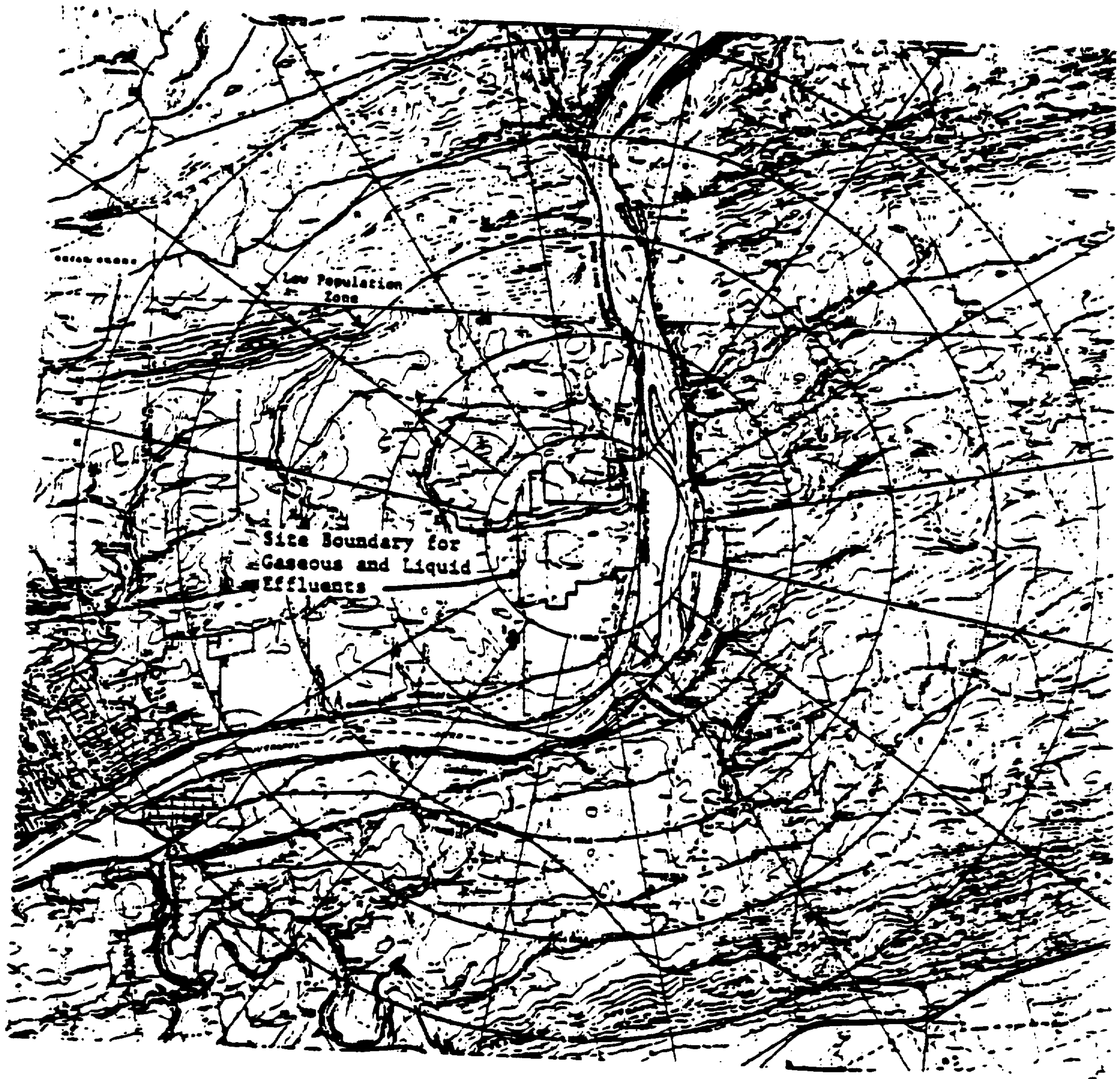


FIGURE 5.1.3-1b

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. One fuel assembly shall contain 78 fuel rods, one inert rod, and 2 water rods. All other fuel assemblies shall contain 79 fuel rods and two water rods. Each fuel rod shall be clad with Zircaloy-2 and have a nominal active fuel length of 150 inches. Reload fuel shall have a maximum average enrichment of 4.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

b. For a pressure of:

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

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

c. For a temperature of 575°F.

VOLUME

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.76 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-388

1.0 INTRODUCTION

By letter dated March 7, 1991 (Ref. 1), the Pennsylvania Power and Light Company and Allegheny Electric Cooperative, Inc. (the licensees) submitted a request for changes to the Susquehanna Steam Electric Station, Unit 2, Technical Specifications (TS). The requested changes would revise the Technical Specifications in support of the ensuing Cycle 5 reload. Changes to the following Technical Specifications and Bases are being made.

B 2.1 Safety Limits

3/4.2.3 Minimum Critical Power Ratio

3/4.4.1 Recirculation System

B 3/4.1 Reactivity Control System

B 3/4.2 Power Distribution Limits

B 3/4.4.1 Recirculation System

5.3.1 Fuel Assemblies

5.3.2 Control Rod Assemblies

2.0 EVALUATION

The Susquehanna 2 Cycle 5 (S2C5) reload will consist of 232 new ANF-4 9x9 fuel assemblies and 532 irradiated ANF 9x9 assemblies. S2C5 will contain no General Electric Company (GE) fuel assemblies. The new 9x9 fuel has similar operating characteristics (mechanical, thermal-hydraulic and nuclear) to the previously used ANF 9x9 reload fuel. In addition to the fuel changes, there will also be a replacement of up to 50 of the current control rod blades with GE designed Duralife 160C blades. In support of the S2C5 reload, the licensee submitted reports which summarize the reload scope (Ref. 2), the plant transient analyses (Ref. 3), and the design and safety analyses (Ref. 4).

Except for the added discussion of the control rod blade replacement the analyses, evaluation and results submitted for S2C5 and the reports referenced are similar to those submitted and approved by the NRC staff for the reload for Cycle 4.

2.1 Fuel Mechanical Design

The S2C5 core reload will include 232 ANF 9x9 fuel bundles with the designation ANF-4. These reload bundles contain 79 fuel rods and 2 water rods. The 232 fuel bundles will have a bundle average enrichment of 3.43 weight percent uranium-235. The fuel design and safety analysis are described in the Susquehanna 2 specific report PL-NF-91-001 (Ref. 2) and the generic mechanical design report XN-NF-85-67(P)(A), Revision 1 (Ref. 5). The NRC has approved the latter report and issued a Safety Evaluation Report on July 23, 1986 (Ref. 6).

Table 2.1 of XN-NF-85-67(P)(A), Revision 1 gives the pertinent design data for ANF 9x9 fuel. Neutronic values specific to the S2C5 reload are given in Table 4.1 of ANF-90-171, Revision 1 (Ref. 4). The burnable poison fuel rods contain 4.0 or 5.0 weight percent gadolinia. The analyses for S2C5 support fuel bundle discharge exposures of 40,000 MWd/MTU. The discharge exposure for this fuel type is 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 S2C5 reload to be acceptable.

For the S2C5 ANF 9x9 reload fuel, calculation of the fuel rod internal pressure was 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 Susquehanna 2 Technical Specifications. This figure was previously approved to reflect the design values which have been reviewed and approved for the ANF 9x9 fuel in connection with the staff's review of XN-NF-85-67(P), Revision 1 (Ref. 5). Based on the results of the generic review, the staff finds the current LHGR limits for the 9x9 fuel to be applicable for the new 9x9 fuel and to be acceptable.

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

2.2 Control Rod Blades

PP&L intends to replace up to 50 of the original equipment control rod blades for S2C5 to meet the commitment, in response to Bulletin 79-26, Revision 1, to limit the B^{10} depletion to no more than 34 percent. The replacement will be General Electric (GE) Duralife 160C blades. They are designed to eliminate B_4C tube cracking and increase blade life. They have improved B_4C tube material, hafnium rods at the blade edge, additional B_4C tubes, increased sheath thickness and other mechanical design improvements. They are about 16 pounds heavier than current Susquehanna blades. With the exception of the improved crevice-free structure and an extended handle, these blades are equivalent to the NRC approved Hybrid I Control Blade Assembly (Ref. 9). The mechanical aspects of the crevice-free structure have been approved by the NRC (Ref. 10). GE has analyzed the blade neutronics using the same methodology as was used for the Hybrid I design. The Duralife 160C blade has a slightly larger reactivity worth than original Susquehanna blades, but it is within the criterion of nuclear interchangeability. The blades weigh less than a D lattice blade (Susquehanna is a C lattice) and the basis of the control rod drop accident drop velocity (which assumes a D lattice rod) remains valid. The scram times associated with the blade are not significantly different than for current blades, and there is a considerable margin to TS scram speed limits. The staff review of these blades concludes that they are acceptable for use in S2C5.

2.3 Nuclear Design

The nuclear design methodology used for S2C5 is that presented in the ANF report XN-NF-80-19(A), Volume 1 and Volume 1 Supplement 1 and 2 (Ref. 11), and the PP&L report PL-NF-87-001-A (Ref. 12), which were reviewed and approved by the staff for application to Susquehanna core reloads.

The minimum value of shutdown margin occurs at 10,125 MWd/MTU and is 1.093 percent delta-k/k. The R-factor is 0.036 percent delta-k/k. Thus the cycle minimum shutdown margin is well in excess of the required 0.38 percent delta-k/k. The Standby Liquid Control System also fully meets shutdown requirements.

The existing new fuel storage calculations are based on k-infinity of the fuel assembly. Based on ANF calculations of 9x9 fuel, an average lattice enrichment of less than 3.95 weight percent uranium-235 and a k-infinity of less than or equal to 1.388 will meet the acceptance criterion of k-effective no greater than 0.95 under dry or flooded conditions. Since the zone average enrichment of the new fuel is 3.43 weight percent uranium-235 and the maximum cold, uncontrolled, beginning-of-life k-infinity for the ANF fuel bundle enriched zones is 1.126, the ANF calculations show that the staff's acceptance criterion is met for the new fuel storage vault under dry and flooded conditions. To preclude criticality at optimum moderation conditions watertight covers and appropriate procedures are used. These are acceptable.

ANF also performed analyses for 9x9 fuel stored in the spent fuel pool. A maximum enriched zone of less than 3.95 weight percent uranium-235 meets the staff acceptance criterion of k-effective no greater than 0.95. Since the ANF-4 9x9 fuel has a zone average enrichment of 3.54 weight percent uranium-235 the staff's acceptance criterion for spent fuel storage is met for the ANF-5 9x9 fuel.

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

2.4 Thermal-Hydraulic Design

The minimum critical power ratio (MCPR) safety limit for the S2C5 reload was determined by the licensee to be 1.06 for all fuel types. The methodology for S2C5 is based on the ANF methodology in XN-NF-80-19(P)(A), Volume 4, Revision 1 (Ref. 13), which has been approved by the staff. The XN-3 correlation used to develop the MCPR safety limit has been approved for the ANF 9x9 fuel (Ref. 14). ANF has determined that this correlation provides sufficient conservatism such that there is no need for any penalty due to channel bow for S2C5. Susquehanna is a C lattice core and uses channels for only one bundle lifetime. For such cores ANF has determined that the conservatism is greater than the maximum expected delta CPR (critical power ratio). The staff has reviewed the ANF channel bow analyses methodology and it is acceptable for this analyses for S2C5.

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. 15). This is used in the MCPR safety limit calculations and as input to the S2C5 transient analyses and is acceptable.

In response to Bulletin 88-07, Supplement 1 (Ref. 16) on BWR thermal-hydraulic stability, PP&L developed restricted operating regions on the power/flow operating map which were in compliance with the NRC recommendations. Technical Specifications (TS) implementing these regions have been approved by the staff for Susquehanna 2. Stability tests have been conducted in Susquehanna 2 with various amounts of ANF 9x9 fuel from succeeding reloads, including all 9x9 fuel. These have indicated no significant deterioration of decay ratio. Decay ratios were low in all tests. Calculations similar to those setting up the restrictive boundaries were done for S2C5. This resulted in slight modifications of the regions for this cycle. TS implementing the changes have been submitted. This review concludes that the analyses are suitable and the changes to the TS are acceptable.

2.5 Transient and Accident Analyses

Various operational transients could reduce MCPR below the safety limit. The most limiting transients have been analyzed to determine which event could potentially result in the largest reduction in the initial Critical Power Ratio (CPR), that is, the delta CPR. The core wide transient which resulted in the largest delta CPR from a 104 percent power and a 100 percent flow condition is the generator load rejection without bypass event (LRWOB). The delta CPR for this event is 0.26. When combined with a safety limit MCPR of 1.06 this results in a MCPR operating limit of 1.32 for S2C5. The most limiting local transient, the control rod withdrawal error (CRWE), was analyzed to support a rod block monitor (RBM) setpoint of 108 percent and resulted in a delta CPR of 0.23. The LRWOB and the CRWE events were the most limiting events for S2C5 at rated power and flow conditions. At less than rated power, the feedwater controller failure (FWCF) event is limiting and a curve of MCPR versus power, which is based on the FWCF results, is included in the Technical Specifications as a power dependent MCPR operating limit.

At reduced flow conditions, the recirculation flow controller failure transient (RFIT) is limiting and MCPR operating limits for manual flow control reduced flow operation for S2C5 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. 17) and the resulting Technical Specification limiting curves are acceptable.

It was assumed for the above analyses that the turbine bypass system and the end-of-cycle recirculation pump trip (RPT) were operable. Analyses were also performed to determine MCPR operating limits with either of these systems inoperable. This resulted in increased MCPR limits which are also proposed for S2C5. 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 S2C5.

Compliance with overpressurization criteria was demonstrated by analysis of the main steam isolation valve (MSIV) closure event assuming MSIV position switch scram failure, an MSIV closure time of 2.0 seconds and six safety-relief valves out-of-service. Maximum vessel pressure was 1,308 psig, within the limit of 1,375 psig. The calculation was done with approved methodology and the results are acceptable.

The LOCA analyses for the Susquehanna plants (Ref. 18) was performed for a full core of ANF 9x9 fuel and is applicable for the S2C5 residual and reload ANF fuel. These analyses have covered an acceptable range of conditions, have been performed with approved methodology and the resulting Technical Specification MAPLHGR values of the ANF fuel remain acceptable.

The control rod drop accident (CRDA) was analyzed with approved ANF methodology (Ref. 11). The maximum fuel rod enthalpy was 213 cal/gm, which is well below the design limit of 280 cal/gm, and less than 253 fuel rods exceed 170 cal/gm, which is less than the 770 rods assumed in the Susquehanna FSAR analysis. To ensure compliance with the CRDA analysis assumptions, control rod sequencing below 20 percent core thermal power must comply with GE's banked position withdrawal sequencing constraints (Ref. 19). The staff concludes that the analysis and results for the S2C5 CRDA are acceptable.

2.6 Single Loop Operation (SLO)

Current Technical Specifications for Susquehanna Unit 2 permit plant operation with a single recirculation loop out-of-service for an extended period of time. Analyses for S2C5 (Ref. 4) show that the MCPR Safety Limit must be increased by 0.01 because of the increased measurement uncertainties. The pump seizure event is more severe under SLO than under two-loop operation, assuming pump seizure of the operating loop. This is the limiting event over most of the power and flow operating region for SLO. ANF analyzed the pump seizure event on a generic basis for the Susquehanna units. Calculations were done for several cycles of operation for the Susquehanna units. The calculated delta CPRs were used to determine a conservative bounding delta CPR. This, combined with a minimum CPR value which would conservatively meet pump seizure accident radiological guidelines of a small fraction of 10 CFR 100 guidelines, resulted in a MCPR operating limit of 1.30 for SLO. This is incorporated as a minimum allowable limit in the Bases to the SLO TS. The flow and power dependent MCPR operating limits given in the TS are always greater than this 1.30 limit for Cycle 5 and, therefore, appropriately protect against the effects of a pump seizure accident under SLO and are acceptable.

Previous analyses reported by the licensee (Refs. 20 and 21) have shown that other events which could be affected by SLO were non-limiting when analyzed under SLO conditions. SLO for S2C5 must maintain the 80 percent recirculation pump speed restriction because of the previous GE vessel internal vibration analysis, as discussed in Reference 20.

2.7 Technical Specification Changes

The following Technical Specification (TS) changes have been proposed for operation of S2C5.

(1) TS 3/4.2.3 -- Figures 3.2.3-1 and -2 are changed to reflect the new calculations of flow and power dependent MCPR operating limits using the parameters of S2C5. As previously discussed, these analyses have been approved and the changes are acceptable.

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

(3) TS 5.3.1 -- This change allows a fuel assembly that failed during Cycle 2 and was returned to use in Cycle 4, with NRC approval, to be retained during Cycle 5. It was repaired by replacing the failed fuel rod with a solid zircaloy-2 rod. This twice-burned, low reactivity assembly will be placed in a non-limiting peripheral region of the core. This change is acceptable.

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

In addition, there are several editorial and descriptive changes to other TS and to the Bases reflecting removal of errors or the reasons for the TS changes discussed above. These include SR 4.4.1.1.2.5, 4.4.1.1.2.6, and 4.4.1.2 (footnote) as well as Bases 2.1.2, 3/4.2.3, and 3/4.4.1. These changes are acceptable.

2.8. Summary

The staff has reviewed the reports submitted for the Cycle 5 operation of Susquehanna Unit 2 and concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area defined in 10 CFR Part 20. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets that eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

Principal Contributors:
Larry Kopp
James Raleigh

Date: April 22, 1991

6.0 REFERENCES

1. Letter (PLA-3533) from H. W. Keiser (PP&L) to W. R. Butler (NRC), "Revised Proposed Amendment 87 to License No. NPF-22: Unit 2 Cycle 5 Reload," March 7, 1991.

2. PL-NF-91-001, "Susquehanna SES Unit 2-Cycle 5: Reload Summary Report," February 1991.

3. ANF-90-170, Revision 1, "Susquehanna Unit 2 Cycle 5: Plant Transient Analysis," February 1991.

4. ANF-90-171, Revision 1, "Susquehanna Unit 2 Cycle 5 Reload Analysis: Design and Safety Analyses," February 1991.

5. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, Inc., September 1986.

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

7. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup - Supplement 1 Extended Burnup Qualification of ENC 9x9 Fuel," May 1988.

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

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

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

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

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

13. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc., June 1986.

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

15. XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," November 1983.

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

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

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

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

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

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