Docket File

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

May 7, 1992

Docket No. 50-387

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

Dear Mr. Keiser:

SUBJECT: CYCLE 7 RELOAD AMENDMENT, SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 (TAC NO. M82356)

The Commission has issued the enclosed Amendment No. 118 to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station, Unit 1. This amendment is in response to your letter dated December 11, 1991.

This amendment changes the Technical Specifications (TS) in support of the Unit 1 Cycle 7 reload. Changes have been made to the following TS and Bases:

Index

B 2.1	Safety Limits
3/4.2.1	Average Planar Linear Heat Generation Rate
3/4.2.2	APRM Setpoints
3/4.2.3	Minimum Critical Power Ratio
3/4.2.4	Linear Heat Generation Rate
3/4.4.1	Recirculation System
B 3/4.1	Reactivity Control Systems
B 3/4.2	Power Distribution Limits
B 3/4.4.1	Recirculation System
5.3.1	Fuel Assemblies

OP-1

DFo1
1/1

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

Enclosures:
1. Amendment No. 118 to License No. NPF-14
2. Safety Evaluation

cc w/enclosures:
See next page

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

Enclosures:
1. Amendment No. 118 to
   License No. NPF-14
2. Safety Evaluation

cc w/enclosures:
See next page

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Pennsylvania Power & Light Company

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Units 1 & 2

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 118
License No. NPF-14

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated December 11, 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-14 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>
   The Technical Specifications contained in Appendix A, as revised through Amendment No. 118 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 and shall be implemented within 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 7, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 118

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

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

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

ATTACHMENT TO LICENSE AMENDMENT NO. 118

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

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

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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 stepback approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specifications 2.1.2 for SNP 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 SNP 9x9 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For the SNP 9x9 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

2.1 SAFETY LIMITS

BASES

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

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

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

2.1.2 THERMAL POWER, High Pressure and High Flow

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

The Safety Limit MCPR assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. XN-NF-524 (A) Revision 1 and PL-NF-90-001 describe the methodologies 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.

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

3/4.2.1 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 limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from 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.

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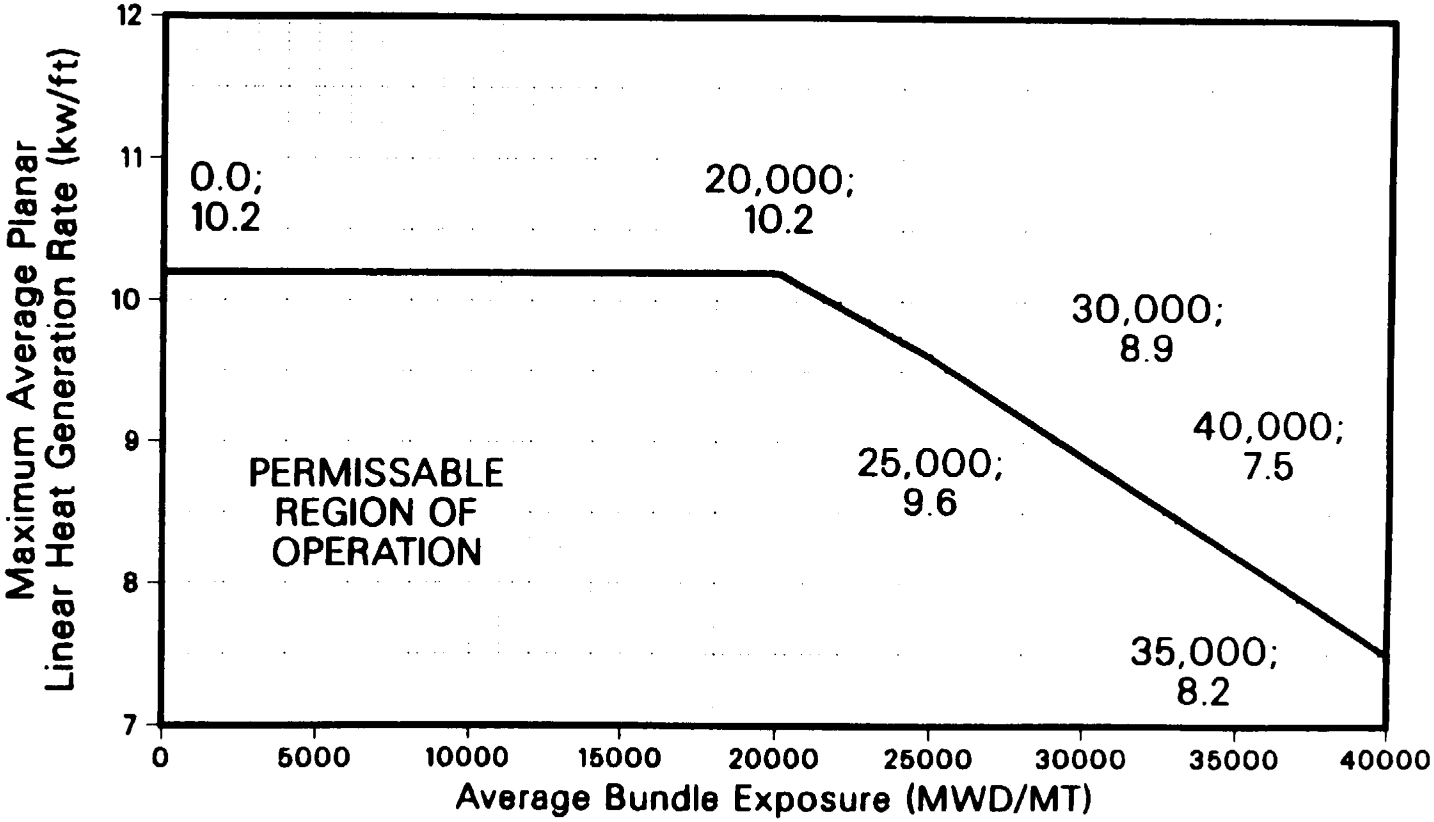
Figure 3.2.1-1 deleted



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

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3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

TRIP SETPOINT *	ALLOWABLE VALUE #
$S \leq (0.58W + 59\%)\ T$ $S_{RB} \leq (0.58W + 50\%)\ T$	$S \leq (0.58W + 62\%)\ T$ $S_{RB} \leq (0.58W + 53\%)\ T$

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

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

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. The FLPD for SNP 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 determined above, initiate corrective action within 15 minutes and adjust S and/ or S_{RB} to be consistent with the Trip Setpoint value* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

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

See Specification 3.4.1.1.2.a for single loop operation requirements.

SURVEILLANCE REQUIREMENTS

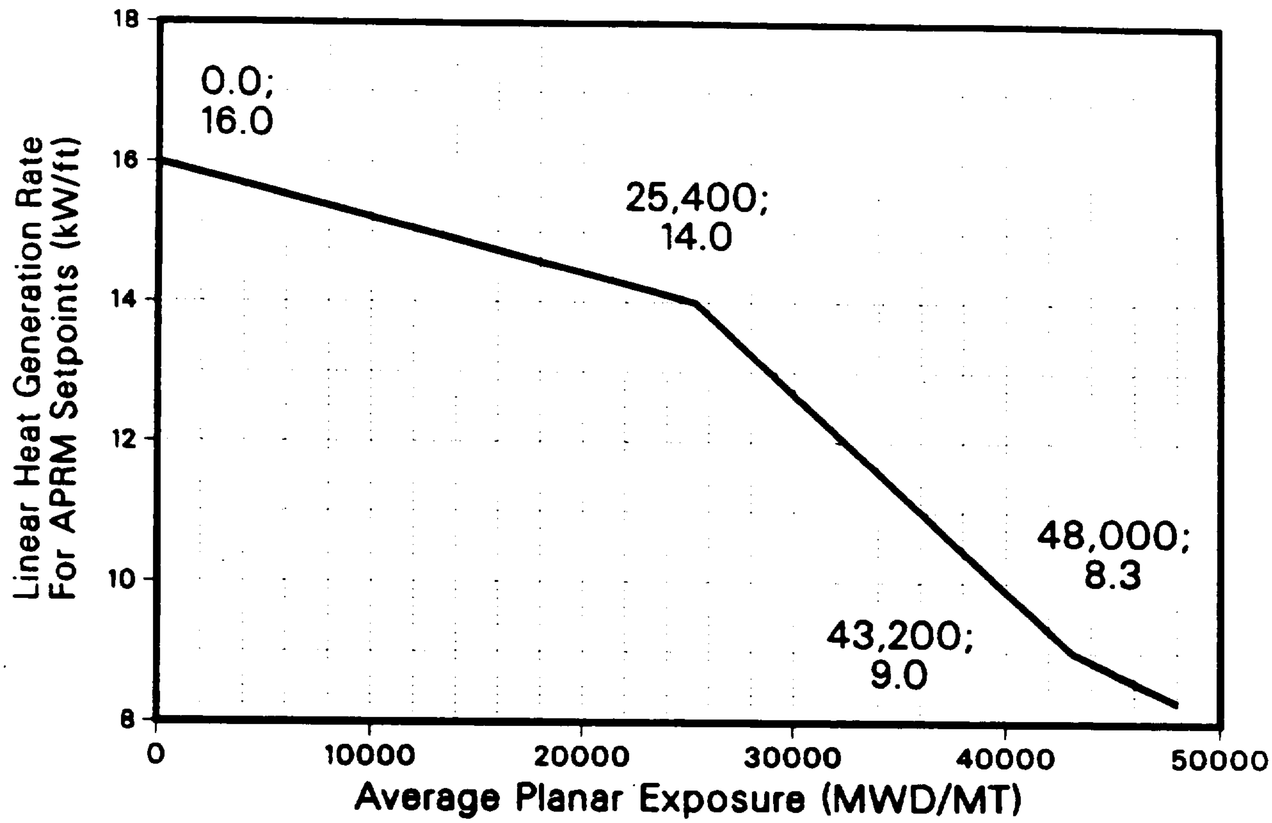
4.2.2 The FRTP and the MFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

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

c. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.



LINEAR HEAT GENERATION RATE FOR APRM SETPOINTS
VERSUS AVERAGE PLANAR EXPOSURE
SNP 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:

a) The Flow-Dependent MCPR value determined from Figure 3.2.3-1, and

b) The Power-Dependent MCPR value determined from one of the following figures, as appropriate:

Figure 3.2.3-2: EOC-RPT and Main Turbine Bypass Operable

Figure 3.2.3-3: Main Turbine Bypass Inoperable

Figure 3.2.3-4: EOC-RPT Inoperable

using a linear interpolation between Curve A and Curve B of the appropriate figure, based on the results of each scram time surveillance test required by Specification 4.1.3.3.

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

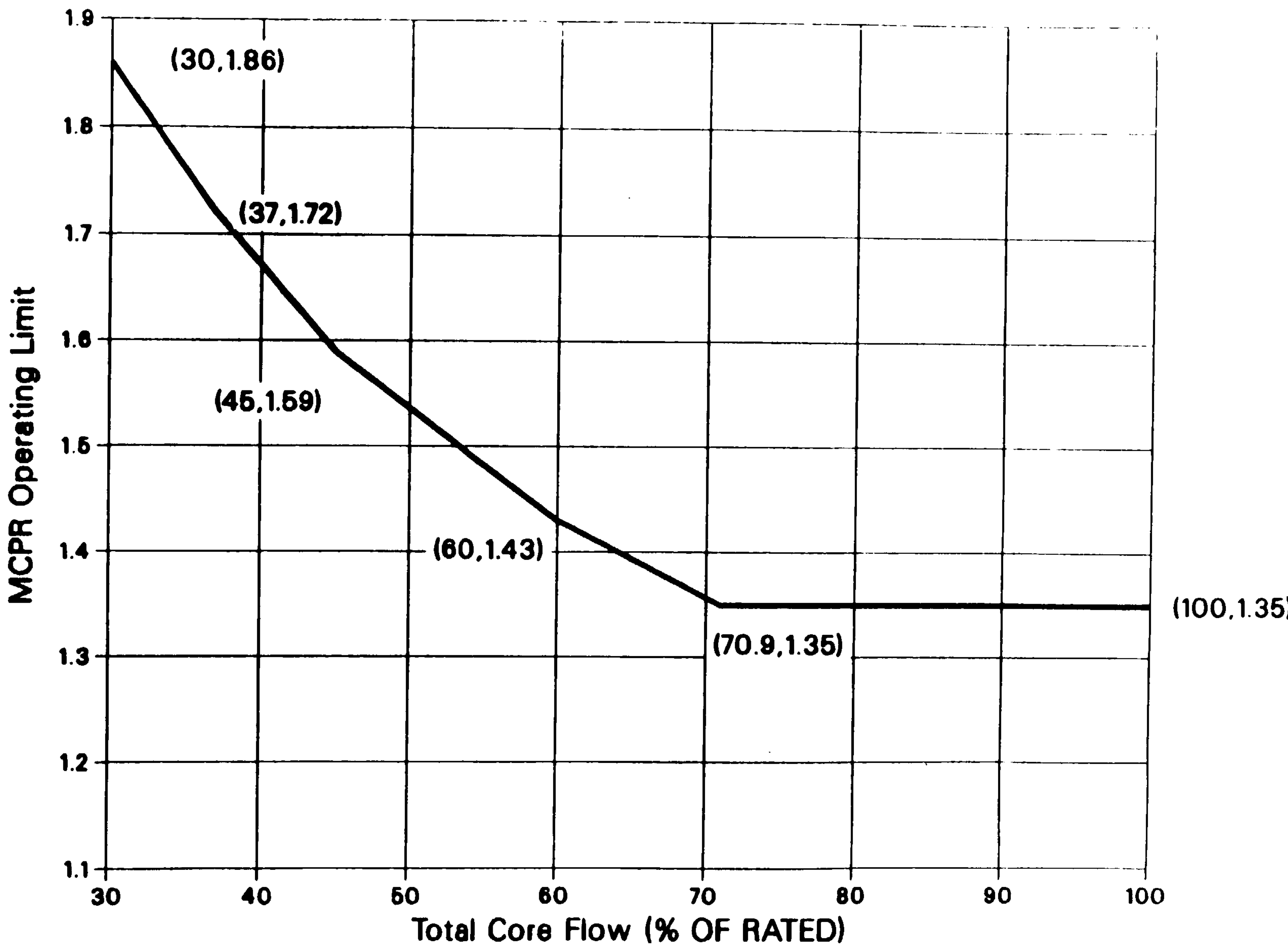
4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit determined from Figure 3.2.3-1 and Figures 3.2.3-2, 3.2.3-3 and 3.2.3-4, as appropriate:

a. At least once per 24 hours,

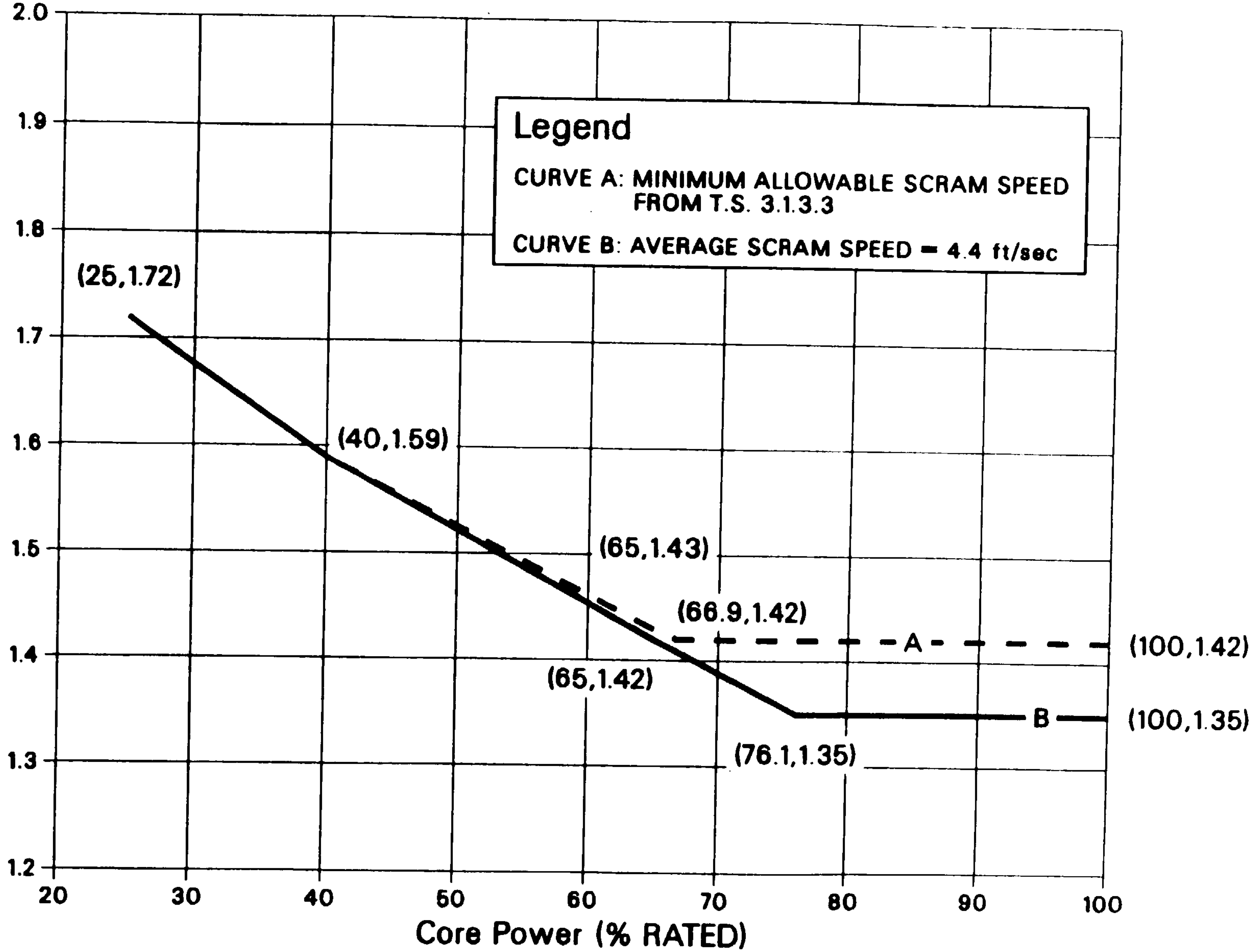
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

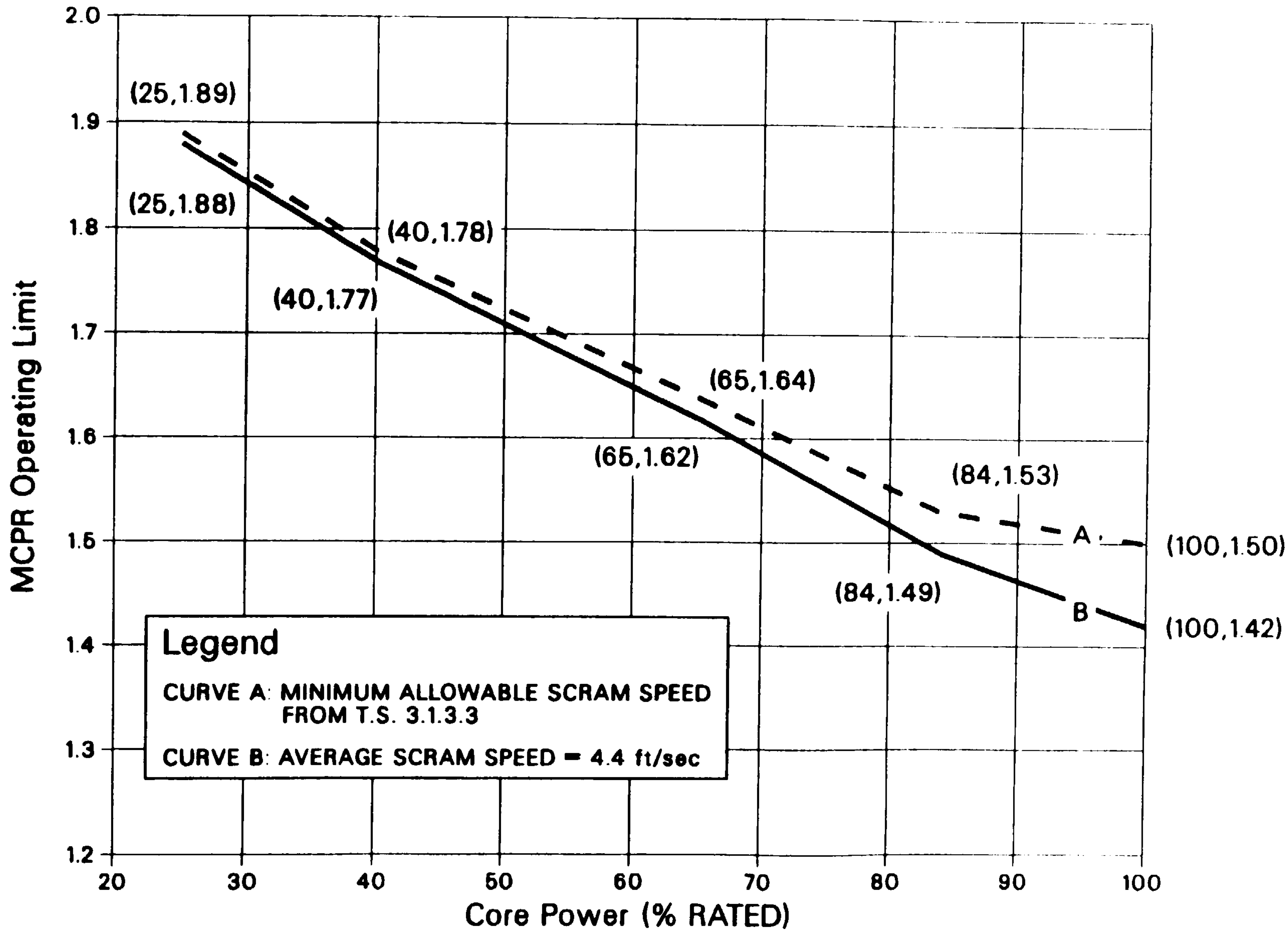
d. The provisions of Specification 4.0.4 are not applicable.



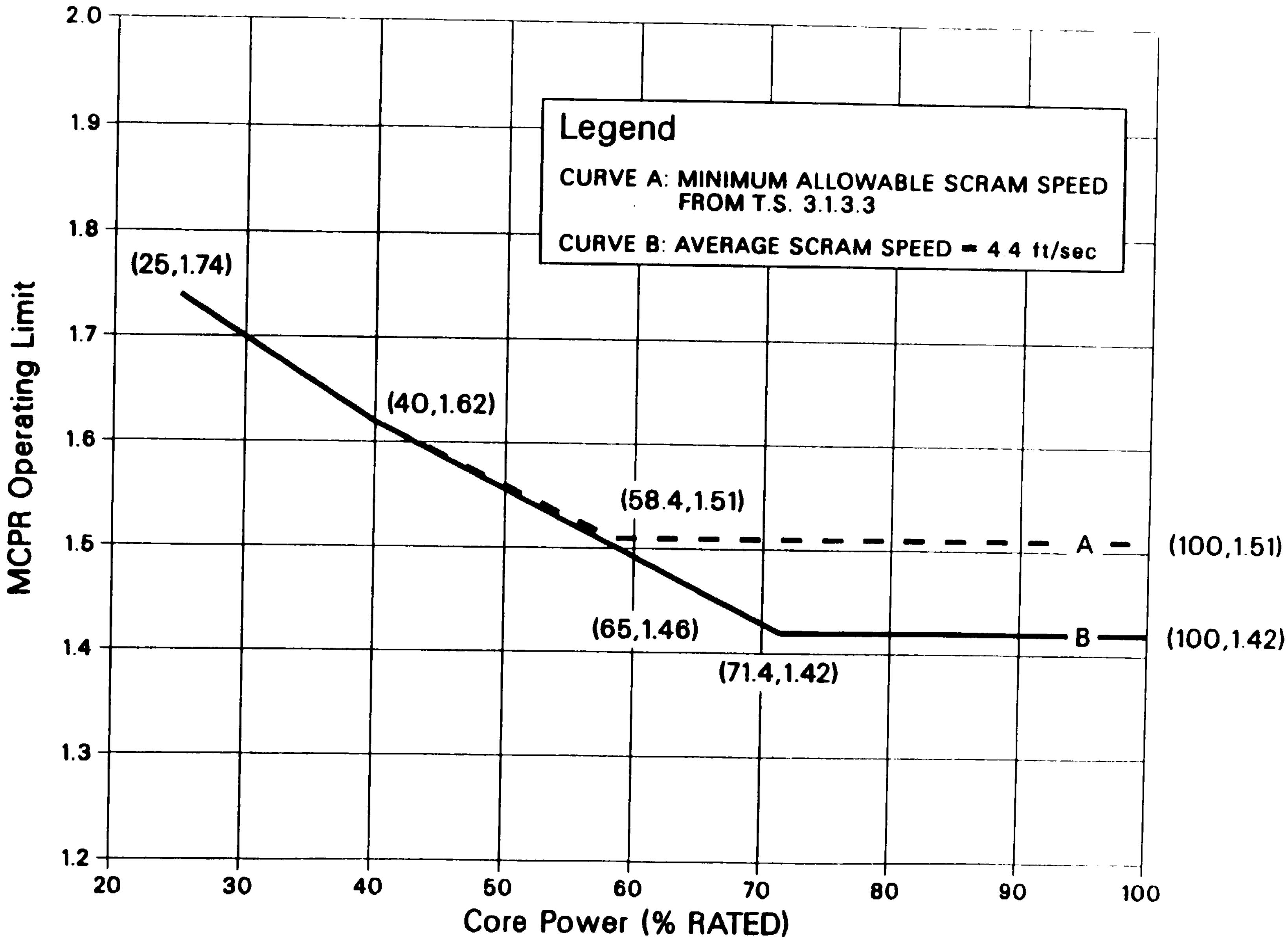
FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1



POWER DEPENDENT MCPR OPERATING LIMIT
EOC-RPT AND MAIN TURBINE BYPASS OPERABLE
FIGURE 3.2.3-2



POWER DEPENDENT MCPR OPERATING LIMIT
MAIN TURBINE BYPASS INOPERABLE
FIGURE 3.2.3-3



POWER DEPENDENT MCPR OPERATING LIMIT
EOC-RPT INOPERABLE
FIGURE 3.2.3-4

3/4.2.4 LINEAR HEAT GENERATION RATE

SNP FUEL

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 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

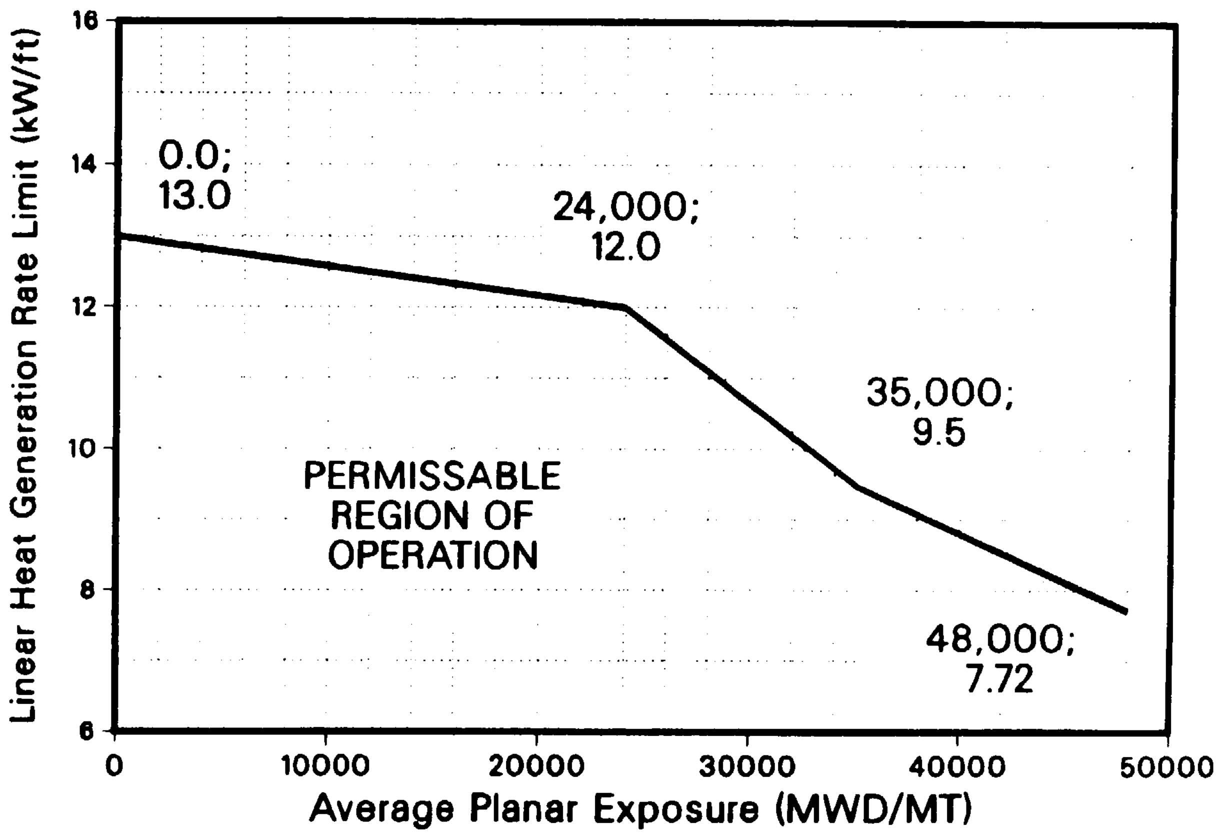
4.2.4 LHGRs shall be determined to be equal to or less than the limit:

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

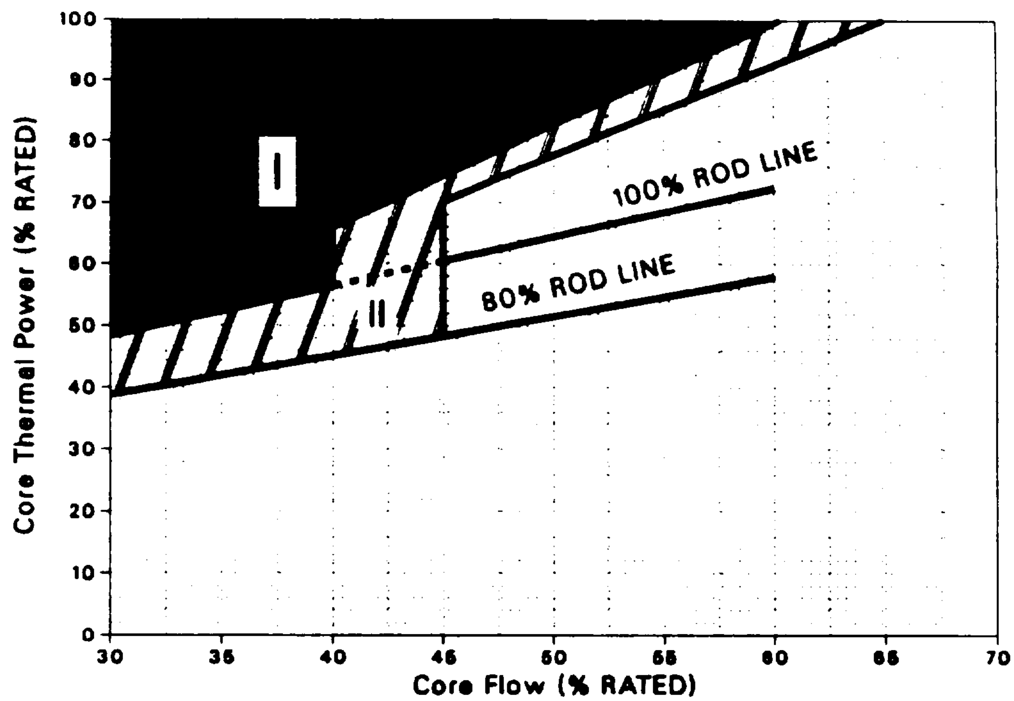
c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

d. The provisions of Specification 4.0.4 are not applicable.



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

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
≤ 0.58W + 54%	≤ 0.58W + 57%

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

Trip Setpoint	Allowable Value
S ≤ (0.58W + 54%) T S_{RB} ≤ (0.58W + 45%) T	S ≤ (0.58W + 57%) T S_{RB} ≤ (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, Figure 3.2.3-3, or Figure 3.2.3-4, as appropriate, 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%
	Trip Setpoint	Allowable Value
b. APRM - Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

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

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%.

** See Specification 4.4.1.1.2.6 for single loop operation requirements.

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta k/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated beginning of cycle shutdown margin minus the calculated minimum shutdown margin in the cycle, where shutdown margin is a positive number. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 Reactivity Anomalies

Since the SHUTDOWN MARGIN requirement is small, a careful check on actual reactor conditions compared to the predicted conditions is necessary. Any changes in reactivity from that of the predicted (predicted core k_{eff}) can be determined from the core monitoring system (monitored core k_{eff}). In the absence of any deviation in plant operating conditions or reactivity anomaly, these values should be essentially equal since the calculational methodologies are consistent. The predicted core k_{eff} is calculated by a 3D core simulation code as a function of cycle exposure. This is performed for projected or anticipated reactor operating states/conditions throughout the cycle and is usually done prior to cycle operation. The monitored core k_{eff} is the k_{eff} as calculated by the core monitoring system for actual plant conditions.

Since the comparisons are easily done, frequent checks are not an imposition on normal operation. A 1% deviation in reactivity from that of the predicted is larger than expected for normal operation, and therefore should be throughly evaluated. A deviation as large as 1% would not exceed the design conditions of the reactor.

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the limit specified in Specification 2.1.2 during the core wide transient analyzed in the cycle specific transient analysis report. The MCPR operating limits in Specification 3.2.3 are a function of average scram speed. Therefore, the results of the required scram time testing (Specification 4.1.3.3) are used to adjust the MCPR operating limits to assure the validity of the cycle specific transient analyses. This ultimately assures that MCPR remains greater than the limit specified in Specification 2.1.2. The occurrence of scram times longer than those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM logic automatically initiates at the low power setpoint (20% of RATED THERMAL POWER) to provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

Parametric Control Rod Drop Accident analyses have shown that for a wide range of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient, effective delayed neutron fraction, and maximum four-bundle local peaking factor are compared with the inputs to the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the

3/4.1.4 CONTROL ROD PROGRAM CONTROLS (Continued)

280 cal/gm design limit to demonstrate compliance for each operating cycle. If cycle-specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle-specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in PL-NF-90-001 and XN-NF-80-19 Volume 1.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum quantity of 4587 gallons of sodium pentaborate solution containing a minimum of 5500 lbs. of sodium pentaborate is required to meet this shutdown requirement. There is an additional allowance of 165 ppm in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel. The temperature requirement for the sodium pentaborate solution is necessary to ensure that the sodium pentaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

BASES

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 SNP 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 ≥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 SNP 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 SNP's Protection Against Fuel Failure (PAFF) line shown in Figure 3.4 of XN-NF-85-67(A), Revision 1. Figure 3.2.2-1 corresponds to the ratio of PAFF/1.2 under which cladding and fuel integrity is protected during AOOs.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR and 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. The required MCPR operating limits as functions core power, core flow, and plant equipment availability condition are presented in Figures 3.2.3-1 through 3.2.3-4.

The transient analyses to determine the MCPR operating limits were performed using methods described in PL-NF-90-001 and corresponding supplements. The pressurization events were analyzed based on a 4.4 ft/sec scram speed as well as the Technical Specification 3.1.3.3 limits. The MCPR operating limits are specified as a function of scram speed.

Figure 3.2.3-1 defines core flow dependent MCPR operating limits which assure that the Safety Limit MCPR will not be exceeded during a flow increase transient resulting from a motor-generator speed control failure. The flow dependent MCPR is only calculated for the manual flow control mode. Therefore automatic flow control operation is not permitted. Figures 3.2.3-2, 3.2.3-3, and 3.2.3-4 define the power dependent MCPR operating limits which assure that the Safety Limit MCPR will not be exceeded in the event of a Feedwater Controller Failure, Rod Withdrawal Error, or Load Reject without Main Turbine Bypass Operable initiated from a full power or reduced power condition.

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

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

MINIMUM CRITICAL POWER RATIO (Continued)

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 fuel rod is less than the design linear heat generation even if fuel pellet densification is postulated.

BASES

3/4.4.1 RECIRCULATION SYSTEM

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

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

The MINIMUM CRITICAL POWER RATIO (MCPR) limits for single loop operation assure that the Safety Limit MCPR is not exceeded for any Anticipated Operational Occurrence (AOO). In addition, the MCPR limits for single-loop operation protect against the effects of the Recirculation Pump Seizure Accident. That is, for operation in single-loop with an operating MCPR limit ≥ 1.30, the radiological consequences of a pump seizure accident from single-loop operating conditions are but a small fraction of 10 CFR 100 guidelines.

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

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

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

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

3/4.4.1 RECIRCULATION SYSTEM (Continued)

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

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

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

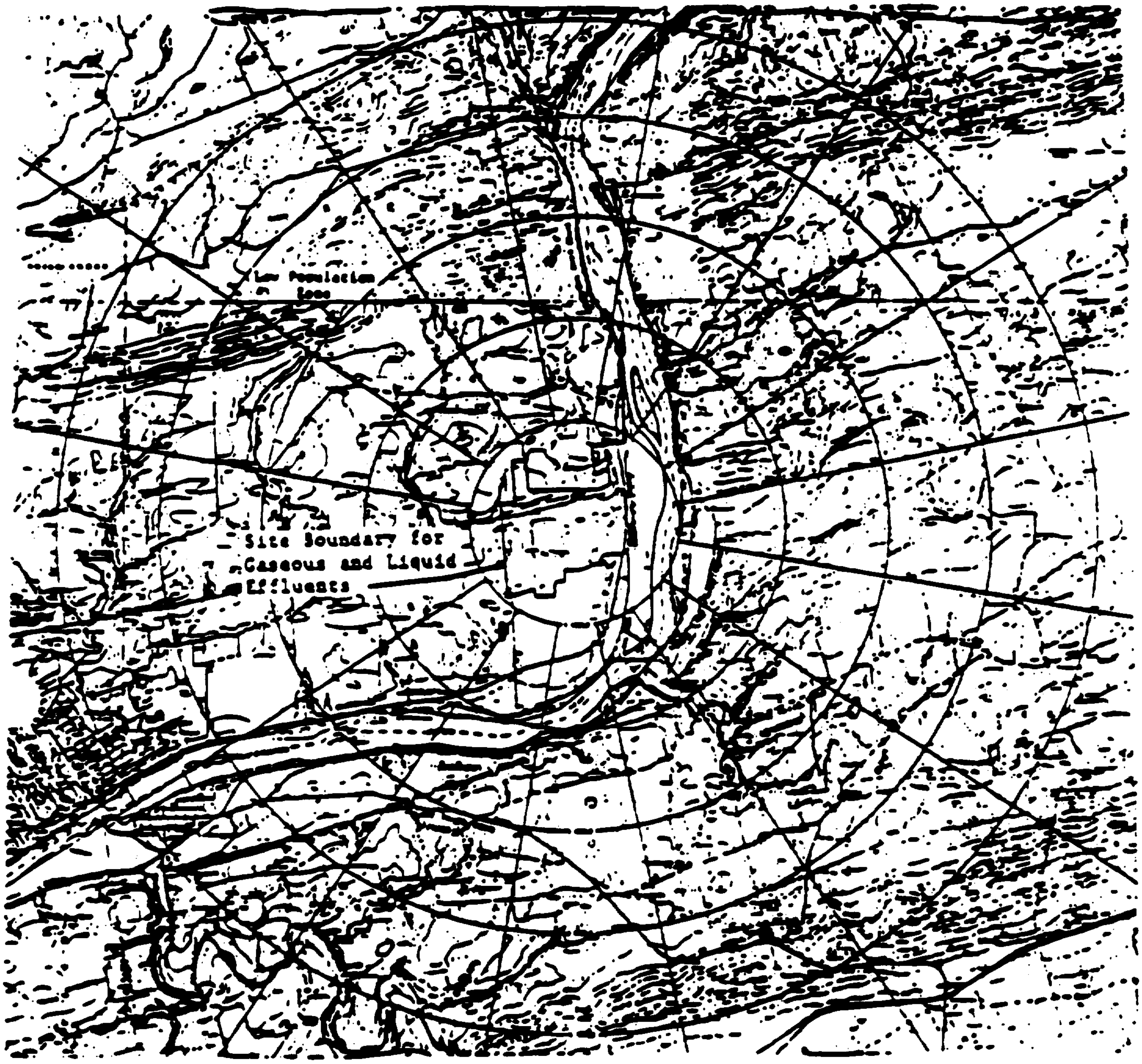


FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED A[illegible]
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. All fuel assemblies shall contain 79 fuel rods and two Zircaloy-2 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) powder near the center of the cruciform, and 143 inch long solid hafnium rods at the edges of the cruciform, all surrounded by a stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

b. For a pressure of:

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

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

c. For a temperature of 575°F.

VOLUME

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NO. NPF-14

PENNSYLVANIA POWER AND LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-387

1.0 INTRODUCTION

By letter dated December 11, 1991, 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 1, Technical Specifications (TS). The requested changes would revise the TS in support of the Unit 1 Cycle 7 reload, changes to the following TS and bases are requested:

Index

B 2.1	Safety Limits
3/4.2.1	Average Planar Linear Heat Generation Rate
3/4.2.2	APRM Setpoints
3/4.2.3	Minimum Critical Power Ratio
3/4.2.4	Linear Heat Generation Rate
3/4.4.1	Recirculation System
B 3/4.1	Reactivity Control Systems
B 3/4.2	Power Distribution Limits
B 3/4.4.1	Recirculation System
5.3.1	Fuel Assemblies

The Susquehanna 1 Cycle 7 (S1C7) reload will consist of 228 fresh (unirradiated) SNP 9X9 (ANF-6) fuel assemblies, 220 once irradiated SNP 9X9 (ANF-5) assemblies, 228 twice irradiated SNP 9X9 (XN-4) assemblies, and 88 thrice irradiated SNP 9X9 (XN-3) assemblies. S1C7 will be the first full core of SNP 9X9 fuel assemblies. The new 9X9 ANF-6 fuel has similar operating characteristics (mechanical, thermal-hydraulic, and nuclear) to the previously used SNP 9X9 reload fuel. In addition to the fuel changes, there will also continue to be replacement of GE original equipment control rod blades with GE designed Duralife 160C blades. In support of the S1C7 reload, the licensee submitted a reload summary report (Ref. 2).

Cycle 7 is the third reload cycle for Unit 1 that PP&L designed by using NRC approved steady state physics methods (Ref. 3). However, this will be the first reload cycle where the licensing analyses were performed by PP&L using safety analysis methods recently approved by the NRC (Refs. 4, 5, 6, 7, 8, 9 and 24). The licensing analyses that PP&L performed are: shutdown margin; standby liquid control system capability; control rod drop accident; loss of feedwater heating; rod withdrawal error; fuel loading error; generator load rejection without bypass; feedwater controller failure; recirculation flow controller failure; and ASME overpressure compliance. SNP provided some supporting analyses including the stability, loss of coolant accident (LOCA), and minimum critical power ratio (MCPR) safety limit analyses. In addition, SNP has previously performed fuel storage criticality, single loop operating, and fuel and equipment handling accident analyses.

2.0 EVALUATION

2.1 Fuel Mechanism Design

The S1C7 core reload will include 228 SNP 9X9 fuel assemblies with the designation ANF-6. These reload assemblies contain 79 fuel rods and 2 water rods in a 9X9 array. Of the 228 ANF-6 fuel assemblies, 156 will contain 9 burnable poison rods with 4.0 weight percent (w/o) Gd_2O_3 and 72 will contain 10 burnable poison rods with 5.0 w/o Gd_2O_3, each blended with UO_2 enriched to 3.40 w/o U-235.

The fuel design and safety analysis are described in the Susquehanna 1 specific report PL-NF-92-007 (Ref. 2) and the generic mechanical design report XN-NF-85-67(P)(A), Revision 1 (Ref. 10). The NRC has approved the latter report and issued a Safety Evaluation Report on July 23, 1986 (Ref. 11).

Table 2.1 of XN-NF-85-67(P)(A), Revision 1, gives the pertinent design data for SNP 9X9 fuel. Neutronic values specific to the S1C7 reload are given in PL-NF-91-007 (Ref. 2). The burnable poison fuel rods contain 4.0 or 5.0 w/o gadolinia. The analyses for S1C7 support fuel assembly discharge exposures of 40,000 MWD/MTU which is based on the approved SNP topical report XN-NF-82-06(P)(A), Supplement 1, Revision 2 (Ref. 12). Based on our review of the information presented, we find the mechanical design of the SNP 9X9 fuel for the S1C7 reload to be acceptable.

Generic analyses were performed by SNP to evaluate the steady state strain, transient strain, cladding fatigue, creep collapse, cladding corrosion, hydrogen absorption, differential fuel rod growth, and grid spacer spring design for the SNP 9X9 fuel design. The RODEX2, RODEX2A, RAMPEX, and COLAPX codes were used in the generic mechanical design analyses. These codes have been approved and/or previously accepted by the NRC and the results indicate that all parameters meet their respective design limits as described in Reference 10.

A figure of linear heat generation rate (LHGR) limit versus planar exposure (MWD/MTU) is incorporated into the Susquehanna 1 Technical Specifications (TS). This figure was previously approved to reflect the design values which have been reviewed and approved for the SNP 9X9 fuel in connection with the staff's review of XN-NF-85-67(P), Revision 1 (Ref. 10). The staff finds the current LHGR limits for the 9X9 fuel to be applicable for the Cycle 7 fuel and to be acceptable.

The licensee has discussed the mechanical response of the SNP 9X9 fuel assembly design during LOCA-seismic events in Reference 2. The discussion compares the physical and structural properties of the SNP 9X9 fuel and the previously used GE 8X8 fuel. The staff has reviewed this information in connection with a previous review and has confirmed that the physical and structural characteristics of the SNP 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, the staff concludes that the original analysis is applicable to S1C7 and the analysis indicating that the design limits are not exceeded is acceptable.

2.2 Control Rod Blades

In response to IE Bulletin 79-26, Rev. 1, PP&L replaced up to 50 of the original equipment rod blades for the previous cycle to meet the commitment to limit the B-10 depletion to no more than 34 percent. The replacement was with 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 original Susquehanna blades. The Duralife 160C control blades have previously been approved by the NRC for use in the previous cycle, S1C6, and are acceptable for use in Cycle 7.

2.3 Nuclear Design

The nuclear design methodology used for S1C7 is that presented in PP&L topical reports PL-NF-87-001-A, PL-NF-89-005, and PL-NF-90-001 (Refs. 3, 4 and 5), and corresponding supplements (Refs. 6, 7, 8, 9 and 24). These reports have been reviewed and approved by the staff for application to Susquehanna core reloads (Ref. 22 and 23).

The minimum value of shutdown margin occurs at beginning of cycle (BOC) and is 1.17% delta k/k. Thus the cycle minimum shutdown margin is well in excess of the required 0.38% delta k/k. The Standby Liquid Control System (SLCS) also fully meets shutdown requirements.

The existing new fuel storage calculations are based on the value of k-infinity of the fuel assembly. Based on SNP calculations of 9X9 fuel, an average lattice enrichment of less than 3.95 weight percent (w/o) U-235 and a k-infinity for the cold (68°F), moderated, uncontrolled fuel assembly lattice in reactor geometry at BOC 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. More recent evaluations of new fuel vault criticality for temperatures as low as 32°F have caused PP&L to reduce their k-infinity criterion to 1.385. Since the zone average enrichment of the new fuel is 3.52 w/o U-235 and the maximum cold, uncontrolled, beginning-of-life (BOS) k-infinity of the two SNP fuel assembly enriched zones are 1.1097 and 1.0749, 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, criticality at optimum moderation conditions, watertight covers, criticality monitors, and appropriate procedures are used. These are acceptable.

SNP also performed analyses for 9X9 fuel stored in the spent fuel pool. A maximum enriched zone of less than 3.95 w/o U-235 meets the staff acceptance criterion of k-effective no greater than 0.95. Since the ANF-6 9X9 fuel has a zone average enrichment of 3.52 w/o U-235, the staff's acceptance criterion for spent fuel storage is met for the ANF-6 9X9 fuel.

Susquehanna will continue to use the SNP 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 2 and has provided acceptable monitoring and predictive results. However, the POWERPLEX input for Cycle 7 will be based on the CPM2/PPL methodology (Ref. 3) reviewed and approved by the NRC. The application of CPM2/PPL generated input in POWERPLEX is described in Reference 5 as supplemented by References 7 and 9, and has also been approved by the NRC. Although the current SNP POWERPLEX power distribution uncertainties were shown to be conservative relative to those obtained by using CPM2 generated input, the NRC has concluded that the safety limit MCPR POWERPLEX uncertainties should remain at their presently approved values when monitoring the core with CPM2/PPL generated input (Ref. 22). The licensee is conforming to this requirement for Cycle 7.

2.4 Thermal-Hydraulic Design

The MCPR for the S1C7 reload was determined by the licensee to be 1.06 for all fuel types. The methodology for S1C7 is based on the SNP methodology in XN-NF-80-19-(P)(A), Volume 4, Revision 1 (Ref. 13), which has been approved by the NRC. The XN-3 critical power correlation used to develop the MCPR safety limit has been approved for the SNP 9X9 fuel. SNP has determined that this correlation provides sufficient conservatism such that there is no need for any penalty due to channel bow for S1C7. Susquehanna is a C lattice core and uses channels for only one fuel bundle lifetime. For such cores, SNP has determined that conservatism is greater than the maximum expected delta CPR (Critical Power Ratio). The staff has reviewed the SNP channel bow analysis methodology and it is acceptable for this analysis for S1C7.

The core bypass flow fraction has been calculated to be 8.4 percent of total core flow using the approved methodology described in PL-NF-87-001-A (Ref. 3). This is used in the MCPR safety limit calculations and as input to the S1C7 transient analyses and is acceptable.

In response to Bulletin 88-07, Supplement 1 (Ref. 14) 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 NRC for Susquehanna 1. Stability tests have been conducted in Susquehanna 2 with various amounts of SNP 9x9 fuel from succeeding reloads, including all 9x9 fuel. These have indicated no significant deterioration of decay ratio. Decay ratios were low in all cases. Calculations similar to those setting up the restrictive boundaries were done for S1C7. 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 transients which resulted in the largest delta CPR were the generator load rejection without bypass (GLRWOB) and the feedwater controller failure (FWCF). These were analyzed based on an average scram speed of 4.4 feet/second and the minimum allowed TS scram speed. Therefore, the power dependent MCPR operating limits for U1C7 are given in the TS as a function of scram speed. The results of the required scram speed time testing (TS 4.1.3.3) will be used to adjust the MCPR operating limits to assure the validity of the Cycle 7 transient analyses. The recirculation flow controller failure (RFCF) event, conservatively analyzed at the TS scram speed, was the limiting event in determining the TS flow dependent MCPR operating limits for Cycle 7. The loss of feedwater heating (LOFWH) event was found to be bounded by these other three core wide transients. The calculations of the thermal margin were performed with approved methodology and the resulting required MCPR operating limits as functions of core power and core flow proposed in TS Figures 3.2.3-1 through 3.2.3-4 are acceptable.

It was assumed for the above analyses that the turbine bypass system and the end-of-cycle (EOC) 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 Cycle 7. These calculations follow standard procedures and operation within the proposed MCPR operating limits with either the main turbine bypass system inoperable or the EOC-RPT inoperable is acceptable for S1C7.

The two limiting local transients were analyzed using the approved methodology described in References 3, 5, 7 and 8). 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 0.22. The fuel loading error, which included analysis of both rotated and mislocated fuel assemblies, was also analyzed and the rotated bundle analysis resulted in the larger delta CPR of 0.22. Both of these events are bounded by the GLRWOB and are, therefore, non-limiting for Cycle 7.

Compliance with the ASME Code overpressurization criterion of 110 percent of vessel design pressure (1375 psig) 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 (current TS minimum closure time is 3.0 seconds), and six safety relief valves out of service. Maximum vessel pressure was 1335.3 psig, within the limit of 1375 psig. The calculation was done with approved methodology and the results are acceptable.

The LOCA analyses for the Susquehanna plants (Ref. 15) was performed by SNP for a full core of SNP 9x9 fuel and is applicable for the S1C7 residual and reload SNP fuel. These analyses have covered an acceptable range of conditions, have been performed with approved methodology, and the resulting TS MAPLHGR values for the SNP fuel remain acceptable.

The control rod drop accident (CRDA) was analyzed with approved PP&L methodology (Ref. 16). The maximum fuel rod enthalpy was 195 cal/gm, which is well below the design limit of 280 cal/gm, and less than 240 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. 17). The staff concludes that the analysis and results for the S1C7 CRDA are acceptable.

2.6 Single Loop Operation (SLO)

Current TS for Unit 1 permit plant operation with a single recirculation loop out-of-service for an extended period of time. However, because of the increased measurement uncertainties, the MCPR safety limit must be increased by 0.01 (Ref. 18). SNP has previously performed SLO analyses for the Susquehanna units (Ref. 19). From these analyses, it was determined that the transients considered for two loop operation bound those for SLO conditions. In addition, it was also determined that postulated accidents under SLO conditions, with the exception of the single loop pump seizure accident, were non-limiting when compared to the postulated accidents under two loop operating conditions. For SLO, it was shown that operation of the Susquehanna units with single loop MCPR operating limits protects against the effects of the pump seizure accident. These analyses have been previously accepted by the NRC (Ref. 20) and are applicable to S1C7. SLO for Cycle 7 continue to maintain the 80 percent recirculation pump speed restriction because of the previous GE vessel internal vibration analysis, as discussed in Reference 21.

2.7 Technical Specification Changes

The following TS changes have been proposed for operation of S1C7.

(1) TS 3/4.2.1 - The changes to this specification are completely editorial in nature in that they reflect that there will be no 8x8 fuel in the Cycle 7 core, and therefore all references to it are being removed. Also, references to "ANF" are updated to "SNP". The changes are acceptable.

(2) TS 3/4.2.2 - The changes to this specification are editorial in nature. They update "ANF" references to "SNP", correct words that were inadvertently reversed, and relocate some figure labels. The changes are acceptable.

(3) 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 S1C7. In addition, Figures 3.2.3-3 and -4 are added. The limits calculated for Cycle 7 will also be a function of scram speed as discussed in Section 2.5 of this SER. As previously discussed, these analyses have been approved and the changes are acceptable.

(4) TS 3/4.2.4 - The changes to this specification are editorial in nature. They delete references to 8x8 fuel, which will not reside in the U1C7 core, and update a reference from "ANF" to "SNP".

(5) TS 3/4.4.1 - Figure 3.4.1.1.1-1 is changed to reflect the calculated changes in the regional stability boundaries. In addition, reference to new Figures 3.2.3-3 and -4 for SLO MCPR operating limits is added. These changes are the result of the MCPR SLO analyses discussed in Section 2.6 of this SER and are acceptable.

(6) TS 5.3.1 - The proposed changes reflect that the Cycle 7 core will contain only 9x9 fuel, and the reference to Zircaloy-2 cladding has been editorially relocated for consistency with the wording in the Unit 2 TS. The changes are acceptable.

In addition, there are several administrative 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 4.4.1.2 (footnote) and Bases 2.1, 3/4.1.3, 3/4.1.4, 3/4.1.5, 3/4.2.1, 3/4.2.2, 3/4.2.3, 3/4.2.4, and 3/4.4.1.

2.8 Fuel Handling Accident

The licensee has revised FSAR chapter 15.7.4 to include an additional fuel handling accident and an equipment handling accident.

In the fuel handling accident, the dropped object is an irradiated fuel assesmbly plus channel, grapple head and mast weighing a total of 1000 pounds which falls from a height of 33 feet above the core. In the equipment handling accident, the dropped object is a mass weighing 1100 pounds which falls from a height of 150 feet above the core. The 33-foot value represents the maximum height that an irradiated fuel assembly can be carried over the core; the 1100 pound mass is the largest object that is not specifically evaluated as a heavy load; and the 150-foot value represents the maximum height that the overhead crane can carry an object over the core.

The number of failed fuel rods for the two cases was determined from the energy of the dropped material and the energy required to fail a fuel rod. The energy required to fail a fuel rod is based upon a uniform 1% plastic deformation of the cladding. For the analysis, the licensee conservatively used the minimum material properties of Zircaloy-2.

For the fuel handling accident analysis, all fuel rods in the dropped assembly are assumed to fail. For the fuel assemblies hit by the dropped material in both analyses, the standard fuel rods and the tie rods are assumed to have the same failure threshold. The energy of the dropped assembly with its fuel handling grapple falling from the vertical position to its side position is included in the calculation. One half of the energy is assumed to be absorbed by the falling fuel assembly and no energy is assumed to be absorbed by the 1100 pound object. For conservatism, no energy is assumed to be absorbed by the fuel pellets. The number of failed fuel rods for the fuel handling accident event is 121; for the equipment handling accident event, the number of failed fuel rods is 318.

The licensee performed the offsite dose calculations assuming (1) the fission product inventories calculated by the ORIGEN computer code are increased by a factor of 1.5, (2) the accident occurs 24 hours after reactor shutdown, (3) the fission gas release fractions are those in Regulatory Guide 1.25, (4) the fuel decontamination factor is 100 for iodine and 1 for noble gases, (5) the standby gas treatment system removal efficiency is 99% for iodine, (6) the atmospheric dispersion factor, breathing rate factor, and dose conversion factors are described in Chapter 1.57.4 of the licensee's FSAR, and (7) the radioactive material is released from the building over a two-hour time period. For the fuel handling accident, the whole body dose calculated by the licensee was 1.3 rem and the thyroid dose was 1.8 rem. For the equipment handling accident, the whole body dose calculated by the licensee was 3.4 rem and the thyroid dose was 4.7 rem.

The licensee has used acceptable methodology to perform these analyses that is based on accident analyses previously submitted and approved by the NRC and on Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors". For each of the two handling accidents analyzed by the licensee, the calculated doses are within the staff's exposure guideline values set forth in Standard Review Plan 15.7.4 (i.e., well within 25% of the 10 CFR 100 values).

2.9 Summary

The staff has reviewed the reports submitted for the Cycle 7 operation of Susquehanna Unit 1 and concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The TS changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

The staff also finds that the proposed changes to the licensee's FSAR chapter 15.7.4 in its December 11, 1991 submittal to be acceptable.

Editorial changes were made to the licensees' incoming Technical Specifications, with the concurrence of the licensee, for clarification. It did not affect the no significant hazards consideration determination.

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 as 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 (57 FR 2598). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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5.0 REFERENCES

1. PLA-3698, "Proposed Amendment 150 to License No. NPF-14: Unit 1 Cycle 7 Reload," December 11, 1991.

2. PL-NF-91-007, "Susquehanna SES Unit 1 Cycle 7 Reload Summary Report," November 1991.

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

4. PL-NF-89-005, "Qualification of Transient Analysis Methods for BWR Design and Analysis," December 21, 1990.

5. PL-NF-90-001, "Application of Reactor Analysis Methods for BWR Design and Analysis," August 1, 1990.

6. PLA-3542, "Response to RAI on PL-NF-89-005," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), March 16, 1991.

7. PLA-3578, "Final Response to RAI on PL-NF-90-001," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), June 4, 1991.

8. PLA-3641, "Licensing Methods: Plan for U1C7," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), August 29, 1991.

9. PLA-3663, "Licensing Methods: Supporting Information," Letter from H. W. Keiser (PP&L) to W. R. Butler (NRC), October 17, 1991.

10. 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.

11. 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.

12. XN-NF-82-06, Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup - Supplement 1 Extended Burnup Qualification of ENC 9x9 Fuel," May 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. NRCB 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)," USNRC Bulletin, December 30, 1988.

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

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

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

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

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

20. Letter from Mohan C. Thadani (NRC) to H. W. Keiser (PP&L), "Cycle 6 Reload, Susquehanna Steam Electric Station, Unit 1 (TAC No. 77165)," November 2, 1990.

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

22. Letter from James Raleigh (NRC) to H. W. Keiser (PP&L), Topical Report PL-NF-90-001, "Application of Reactor Analysis Methods for BWR Design and Analysis," Susquehanna Steam Electric Station, Units 1 and 2, November 21, 1991.

23. Letter from James Raleigh (NRC) to H. W. Keiser (PP&L), Topical Report PL-NF-89-005, "Qualification of Transient Analysis Methods for BWR Design and Analysis," Susquehanna Steam Electric Station, Units 1 and 2, March 1992.

24. PLA-3729, "Response to RAI on Transient Analysis Methods," Letter from H. W. Keiser (PP&L) to C. L. Miller (NRC), February 12, 1992.