SUSQUEHANNA SES UNIT 1 CYCLE 6

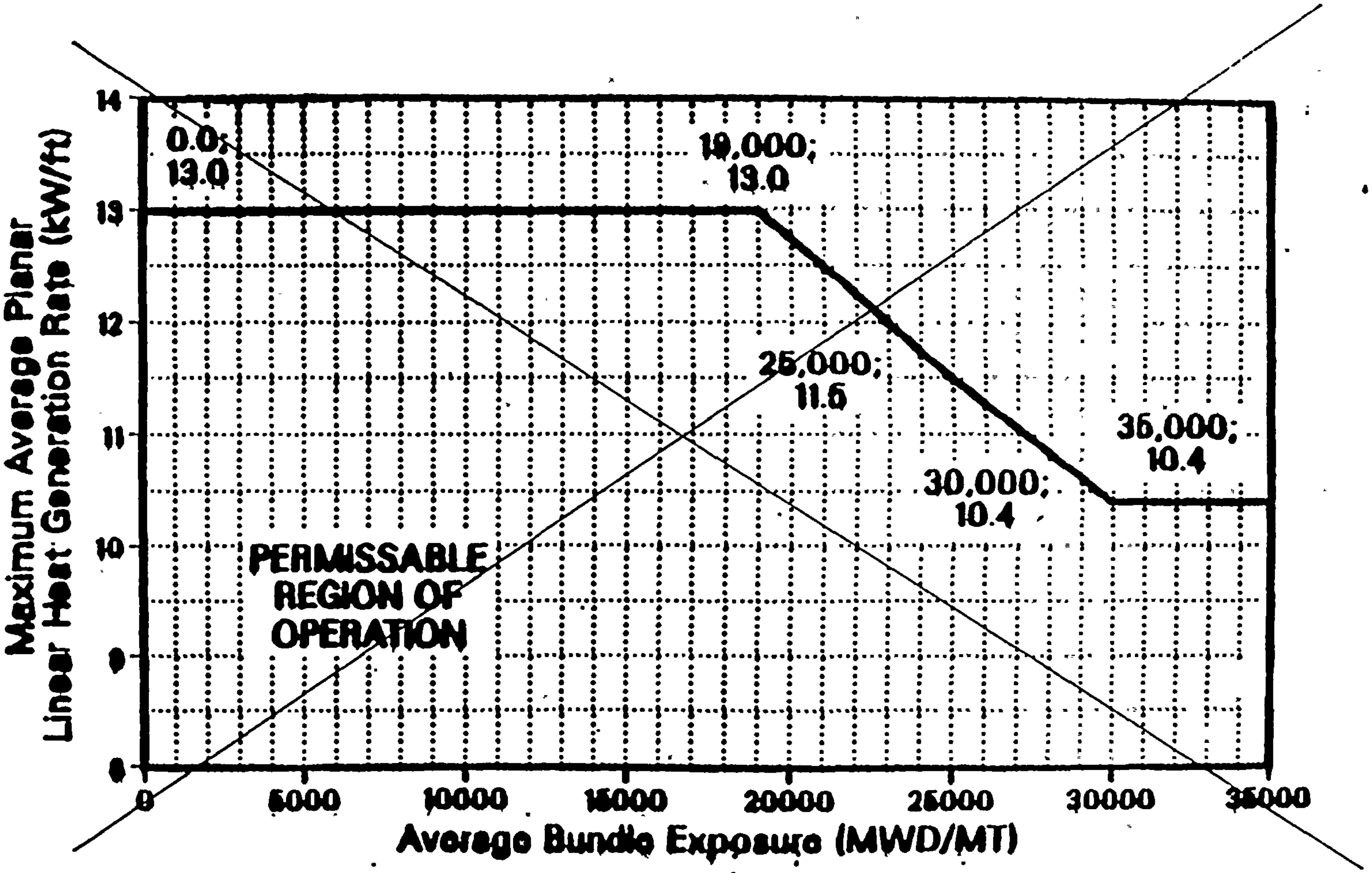
TECHNICAL SPECIFICATION CHANGES

June 1990

PENNSYLVANIA POWER & LIGHT COMPANY

PP&L

LIST OF FIGURES



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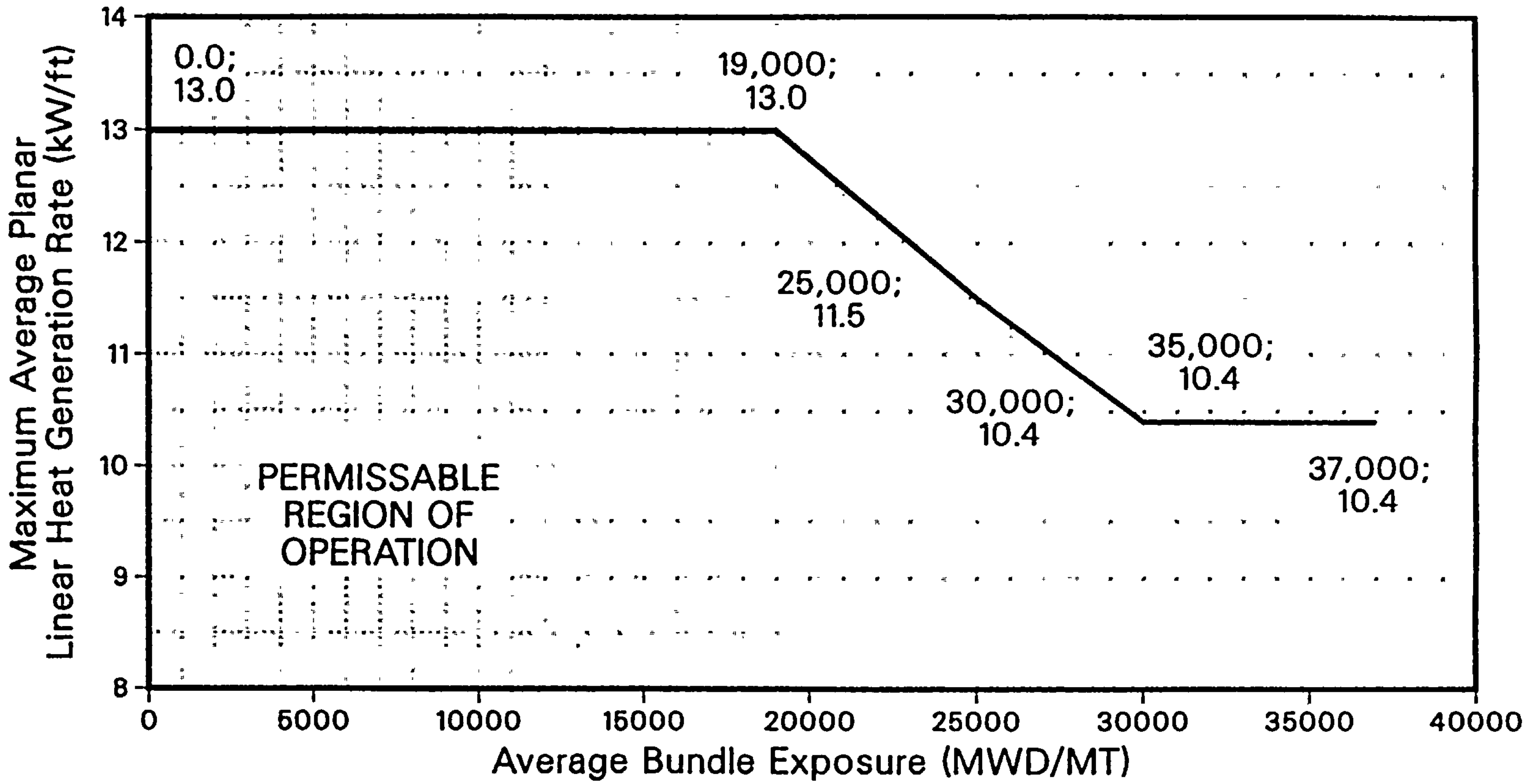
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
ANF 8X8 FUEL
FIGUR` 3.2.1-1



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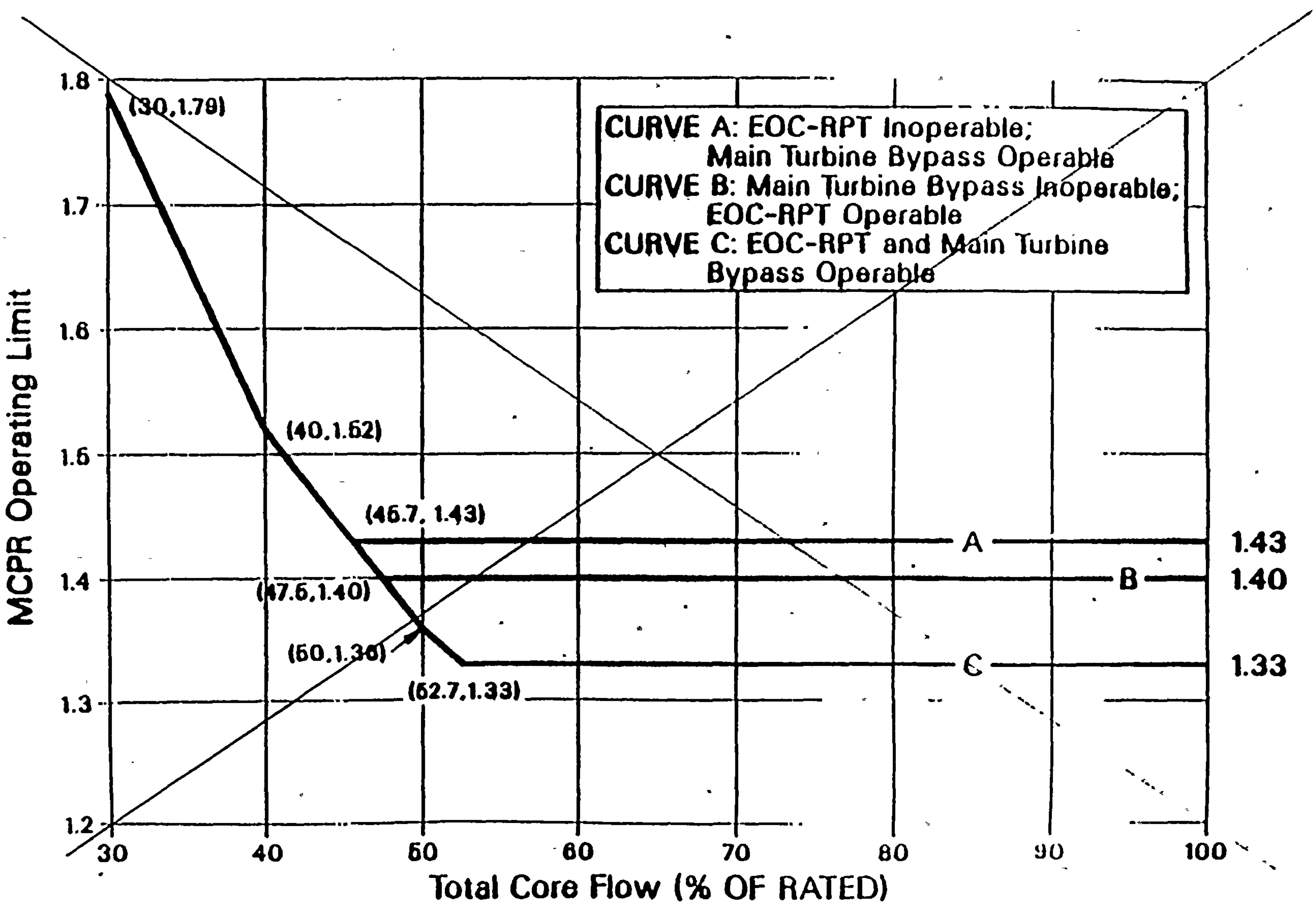
ANF 8X8 FUEL
FIGURE 3.2.1-1



FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1

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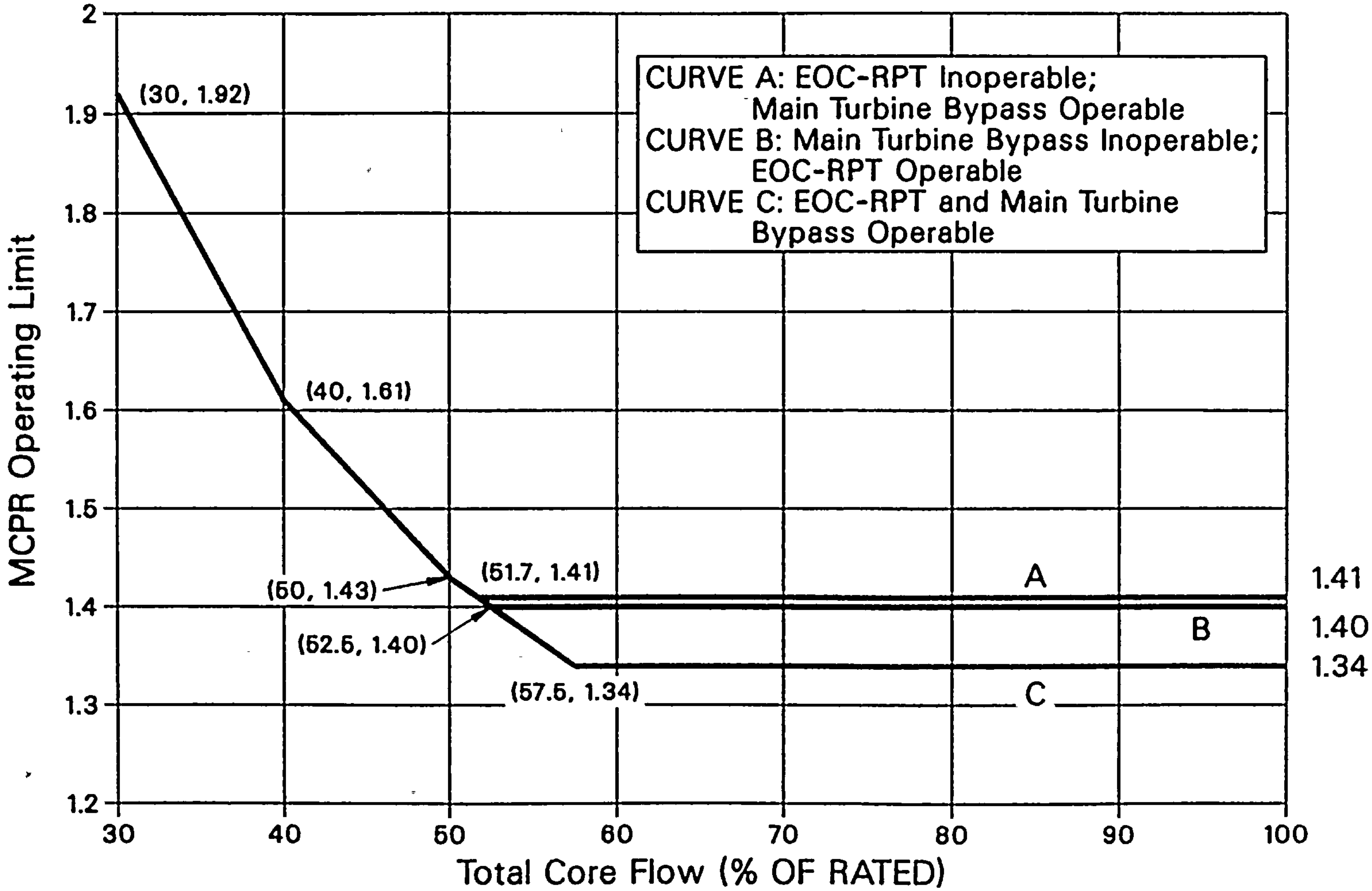
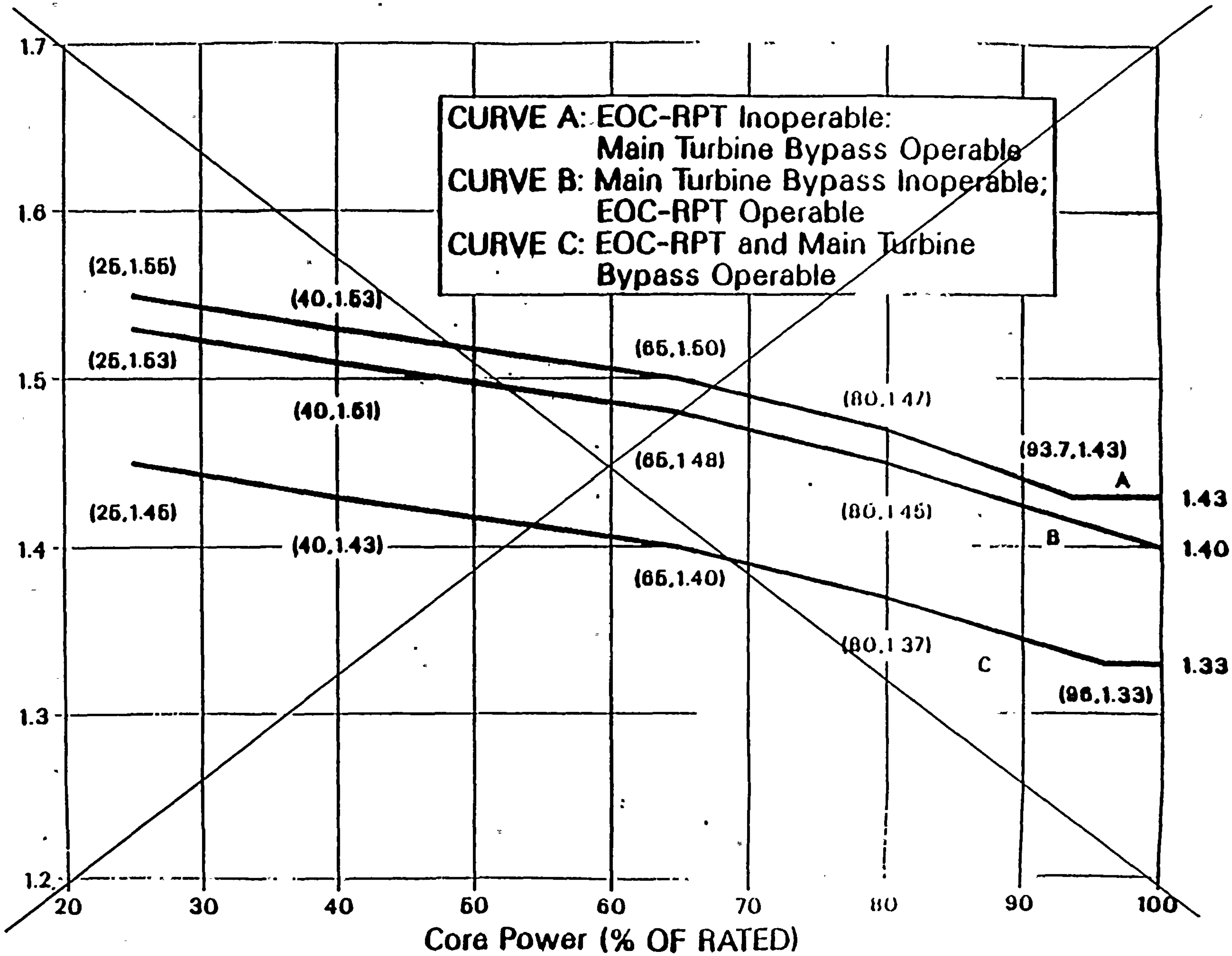


FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1



REDUCED POWER MCPR OPERATING LIMIT

Figure 3.2.3-2

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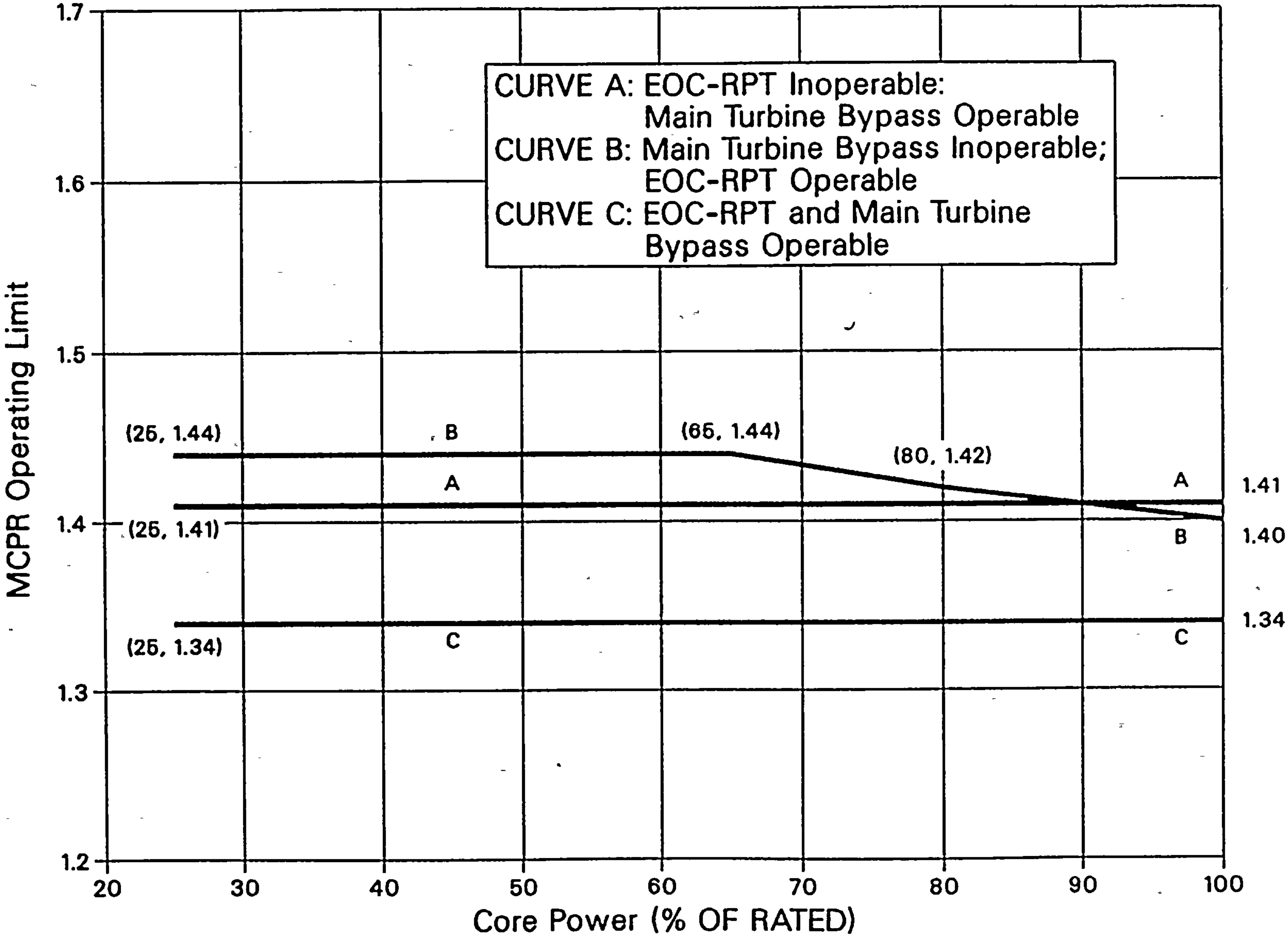
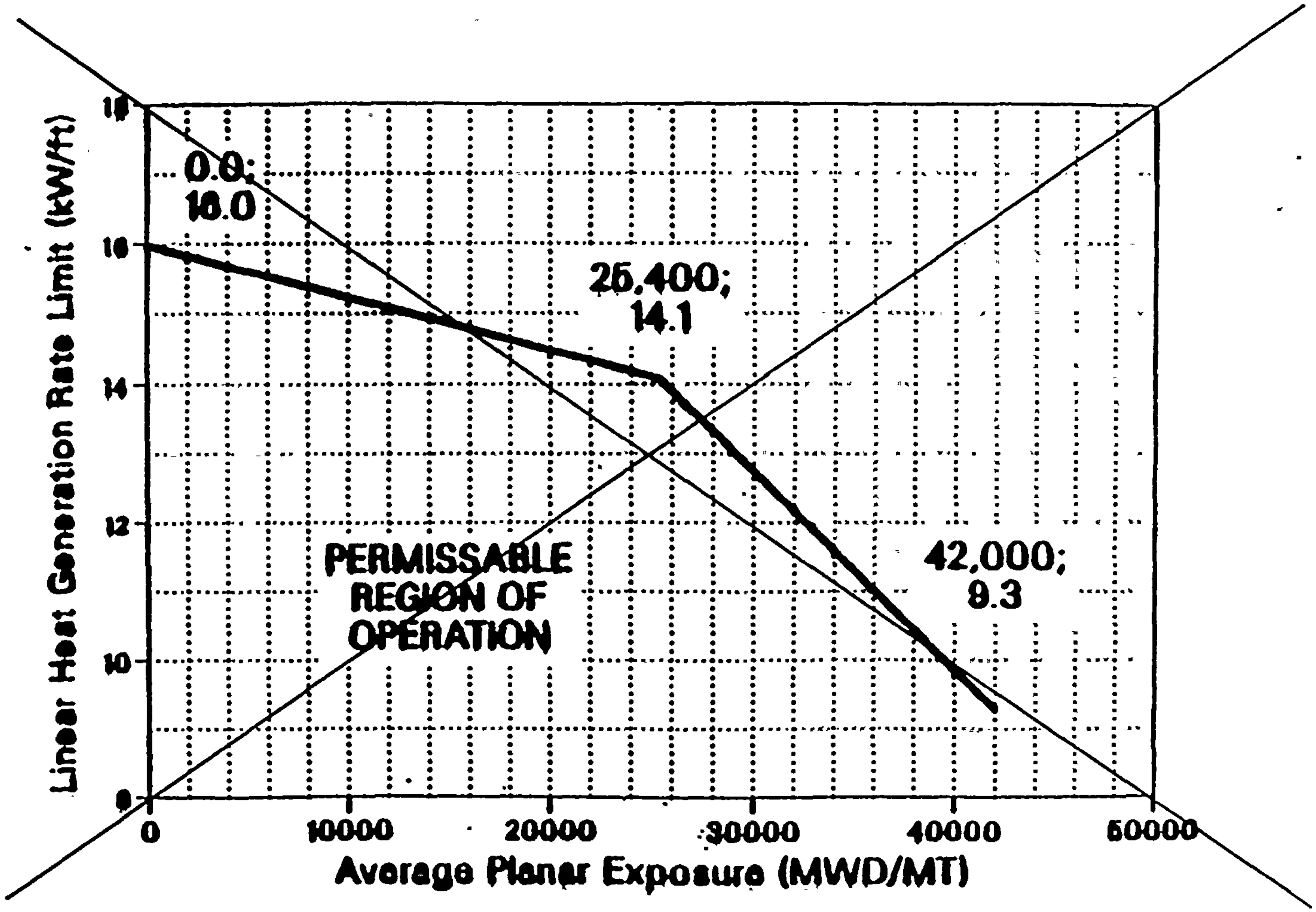


REDUCED POWER MCPR OPERATING LIMIT
Figure 3.2.3-2



LINEAR HEAT GENERATION RATE (LHGR) LIMIT
VERSUS AVERAGE PLANAR EXPOSURE
ANF 8X8 FUEL
FIGURE 3.2.4-1

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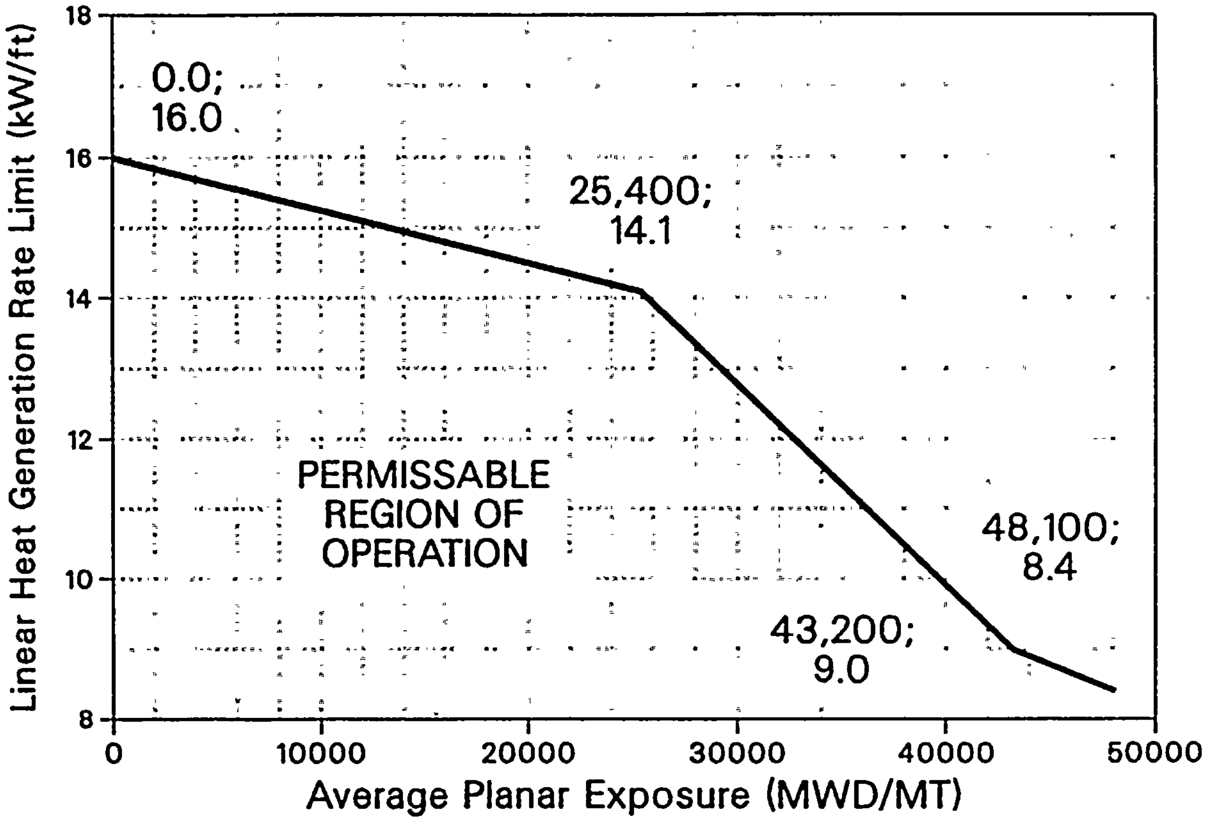
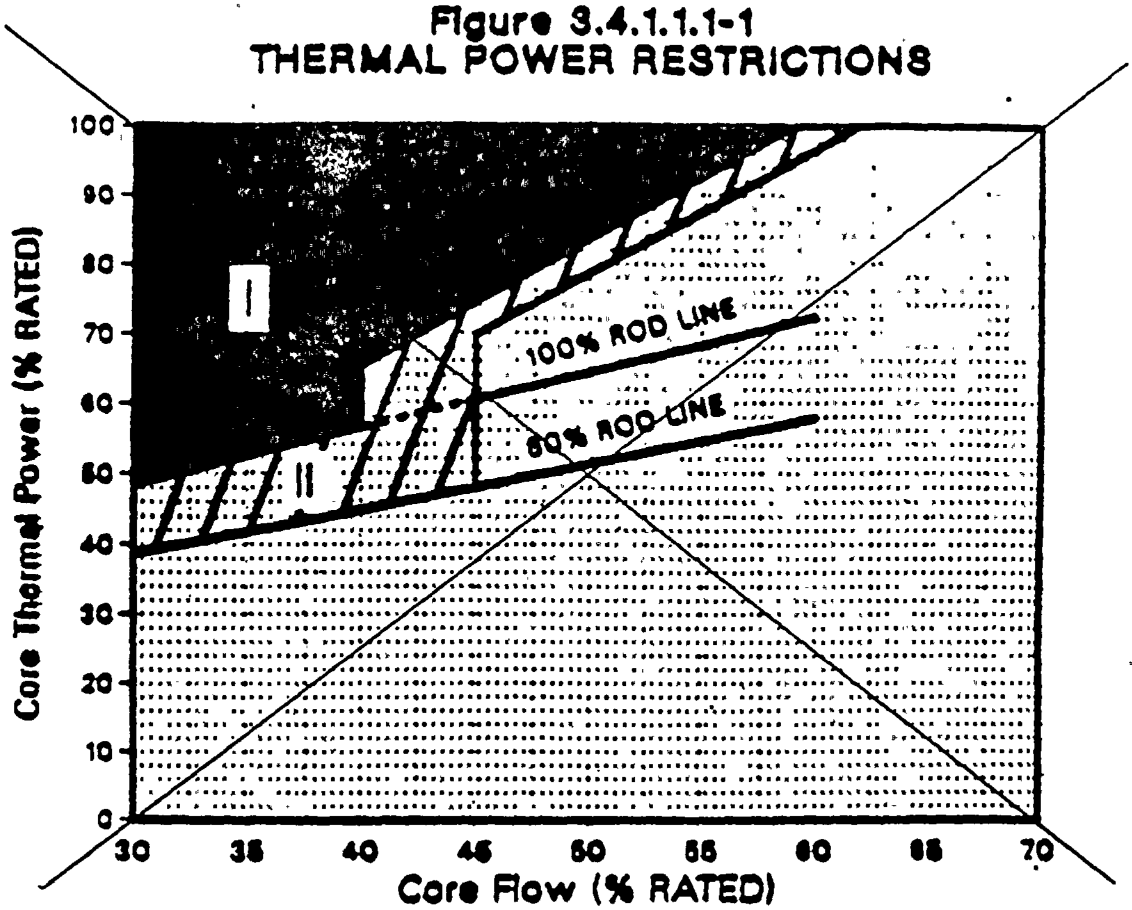


LINEAR HEAT GENERATION RATE (LHGR) LIMIT
VERSUS AVERAGE PLANAR EXPOSURE
ANF 8X8 FUEL
FIGURE 3.2.4-1

Figure 3.4.1.1.1-1
THERMAL POWER RESTRICTIONS



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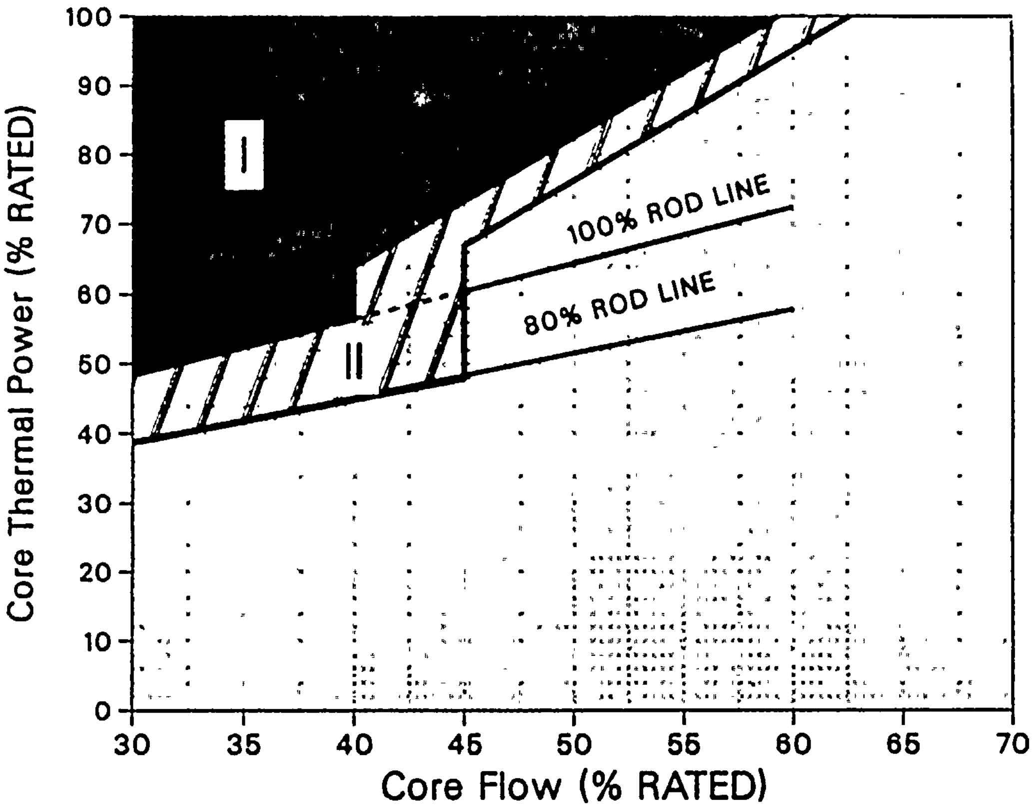
~~FIGURE 3.4.1.1-1~~
~~THERMAL POWER RESTRICTIONS~~



THERMAL POWER RESTRICTIONS
Figure 3.4.1.1.1-1

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 ≤ (0.58W + 57%)T
S_{RB} ≤ (0.58W + 45%)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. ~~1.42,~~ → 1.30

   b. the MCPR determined from Figure 3.2.3-1 plus 0.01, and

   c. the MCPR determined from Figure 3.2.3-2 plus 0.01.

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

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

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

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 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 POWER DISTRIBUTION LIMITS

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The Technical Specifiation APLHGR for ANF fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is shown in Figures 3.2.1-1 and 3.2.1-2.

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

X

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analyses. In addition, the APRM setpoints must be adjusted to ensure that $\geq$1% plastic strain and fuel centerline melting do not occur during the worst anticipated operational occurrence (AOO), including transients initiated from partial power operation.

For ANF fuel the T factor used to adjust the APRM setpoints is based on the FLPD calculated by dividing the actual LHGR by the LHGR obtained from Figure 3.2.2-1. The LHGR versus exposure curve in Figure 3.2.2-1 is based on ANF's Protection Against Fuel Failure (PAFF) line shown in Figure 3.4 of XN-NF-85-67(A), Revision 1. Figure 3.2.2-1 corresponds to the ratio of PAFF/1.2 under which cladding and fuel integrity is protected during AOOs.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figures 3.2.3-1 and 3.2.3-2.

The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to an ANF core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and non-pressurization events are described in XN-NF-79-71 and XN-NF-84-105. The principal result of this evaluation is the reduction in MCPR caused by the transient.

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

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)

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

3/4.2.4 LINEAR HEAT GENERATION RATE

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

~~References:~~

~~1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.~~

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.

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)~~ and for the Recirculation Pump Seizure Accident~~.

X

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 3.4.1.1.1.1-1 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.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 or 79 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches. ~~The initial core loading shall have a maximum average enrichment of [illegible] weight percent U-235.~~ Reload fuel ~~shall be similar in physical design to the initial core loading and~~ shall have a maximum average enrichment of 4.0 weight percent U-235.

Replace with "Insert A" as provided on the following page.

CONTROL ROD ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

b. For a pressure of:

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

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

c. For a temperature of 575°F.

VOLUME

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

Insert A

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.

NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following three questions are addressed for each of the proposed Technical Specification changes:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

3. Does the proposed change involve a significant reduction in a margin of safety?

Specification 3/4.2.1, Average Planar Linear Heat Generation Rate

The changes to this specification are solely to Figure 3.2.1-1, which provides appropriate MAPLHGR limits to bound the exposure that the ANF 8x8 fuel will experience during Cycle 6 operation.

1. No. The increased allowed exposure is based on an additional MAPLHGR evaluation performed by ANF (See Summary Report Reference 3). This evaluation is consistent with previously approved methods, and ensures that the peak cladding temperature for the ANF 8x8 fuel remains below 2200°F, local Zr-H_2O reaction remains below 17%, and core-wide hydrogen production remains below 1% for the limiting LOCA event as required by 10 CFR 50.46. Therefore, the additional MAPLHGR limits do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The analysis described above can only be evaluated for its effect on the consequences of analyzed events; it cannot create new ones. The consequences of analyzed events were evaluated in 1. above.

3. No. As discussed in 1. above, the analysis to support the MAPLHGR limits at higher exposures is consistent with previously approved methods and meets all pertinent regulatory criteria for use in this application. Therefore, the proposed change will not result in a significant decrease in any margin of safety.

Specification 3/4.2.3, Minimum Critical Power Ratio

The changes to this specification provide new operating limit MCPR curves based on cycle-specific transient analyses.

1. No. Limiting core-wide transients were evaluated with ANF's COTRANSA code (see Summary Report Reference 29) and this output was utilized by the XCOBRA-T methodology (see Summary Report Reference 30) to determine delta CPRs. Both COTRANSA and XCOBRA-T have been approved by the NRC in previous license amendments. All core-wide transients were analyzed deterministically (i.e., using bounding values as input parameters).

Two local events, Rod Withdrawal Error and Fuel Loading Error, were analyzed in accordance with the methods described in XN-NF-80-19 (A) Vol. 1 (see Summary Report Reference 6). This methodology has been approved by the NRC.

Based on the above, the methodology used to develop the new operating limit MCPRs for the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The methodology described can only be evaluated for its effect on the consequences of analyzed events; it cannot create new ones. The consequences of analyzed events were evaluated in 1. above.

3. No. As stated in 1. above, and in greater detail in the attached Summary Report, the methodology used to evaluate core-wide and local transients is consistent with previously approved methods and meets all pertinent regulatory criteria for use in this application.

   Based on the above, the use of the methodology utilized to produce the U1C6 MCPR operating limits will not result in a significant decrease in any margin of safety.

Specification 3/4.2.4, Linear Heat Generation Rate

Proposed changes to this specification provide appropriate limits at extended burnups for ANF 8x8 fuel.

1. No. ANF-90-018(P), Revision 1 (see Summary Report Reference 5) supports the new maximum 8X8 discharge exposure. This report demonstrates that margin to 8x8 fuel mechanical design limits is assured for all anticipated operational occurrences throughout the life of the fuel provided that the fuel rod power history remains within the power histories assumed in the analyses.

   Based on the above, the U1C6 LHGR operating limits do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. This change reflects appropriate limits which ensure compliance with all relevant fuel mechanical design criteria. Application of these limits will not create the possibility of a new or different event.

3. No. As described in 1. above, ANF-90-018(P) Revision 1 demonstrates appropriate safety margin to fuel mechanical design limits for all anticipated operational occurrences throughout the life of the fuel.

Specification 3/4.4.1, Recirculation System (Two Loop Operation)

The changes to this specification (i.e., Figure 3.4.1.1.1-1) reflect cycle-specific stability analysis.

1. No. COTRAN core stability calculations were performed for Unit 1 Cycle 6 to determine the decay ratios at predetermined power/flow conditions.

The resulting decay ratios (See Summary Report, Reference 3) were used to define operating regions which comply with the interim requirements of NRC Bulletin No. 88-07, Supplement 1 "Power Oscillations in Boiling Water Reactors," (See Summary Report, Reference 19). As in the previous cycle, Regions B and C of the NRC Bulletin have been combined into a single region (i.e., Region II), and Region A of the NRC Bulletin corresponds to Region I.

Region I has been defined such that the decay ratio for all allowable power/flow conditions outside of the region is less than 0.90. To mitigate or prevent the consequences of instability, entry into this region requires a manual reactor scram. Region I for Unit 1 Cycle 6 has been calculated to be slightly larger than Region I for the previous cycle.

Region II has been defined such that the decay ratio for all allowable power/flow conditions outside of the region (excluding Region I) is less than 0.75. For Unit 1 Cycle 6, Region II must be immediately exited if it is inadvertently entered. Similar to Region I, Region II is slightly larger than in the previous cycle.

In addition to the region definitions, PP&L has performed stability tests in SSES Unit 2 during initial startup of Cycles 2 and 3 to demonstrate stable reactor operation with ANF 9x9 fuel. The test results for U2C2 (See Summary Report, Reference 20) show very low decay ratios with a core containing 324 ANF 9x9 fuel assemblies.

Figure 3/4.1.1.1-1 is also referenced by Specification 3/4.4.1.1.2, which governs Single Loop Operation (SLO). The evaluation above applies under SLO conditions as well.

Based on the above, operation within the limits specified by the proposed changes will ensure that the probability and consequences of unstable operation will not significantly increase.

2. No. The methodology described above can only be evaluated for its effect on the consequences of unstable operation; it cannot create new events. The consequences were evaluated in 1. above.

3. No. PP&L believes that the use of Technical Specifications that comply with NRC Bulletin 88-07 Supplement 1, and the tests and analyses described above, will provide assurance that SSES Unit 1 Cycle 6 will comply with General Design Criteria 12, Suppression of Reactor Power Oscillations. This approach is consistent with the SSES Unit 1 Cycle 5 method for addressing core stability (See Summary Report, References 22 and 23).

<u>Specification 3/4.4.1, Recirculation System (Single Loop Operation)</u>

The changes to this specification include a revised MCPR limit and correction of a typographical error.

1. No. The revised MCPR limit reflects the result of ANF's analysis of a recirculation pump seizure accident on a generic basis for the

Susquehanna units (See Summary Report Reference 4). Past analyses of this accident utilized ANF's transient methodology to establish a delta CPR which would preclude fuel failures due to overheating or clad strain. The generic analysis performed for U1C6 and future SSES cycles used Safety Limit MCPR methodology to determine the extent of rods which might experience boiling transition should MCPR reach 0.90. This accident methodology results in increased consequences (less than 2% of the fuel rods were calculated to experience boiling transition at the 95% confidence level, and significantly fewer rods would be expected to fail, as opposed to none using the transient methods). This result, however, is not a significant increase in consequences when compared to LOCA results. Furthermore, it meets the regulatory acceptance criteria for radiological consequences since they are but a small fraction of 10 CFR 100 guidelines, even with the conservative assumption that all rods which experience boiling transition fail.

The typographical error is an inadvertent omission of the "W" in the APRM flow biased trip setpoint. This is an editorial correction to a previously approved amendment; no technical change is being proposed.

Based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The analysis which supports the SLO MCPR limit revision can only be evaluated for its effect on the consequences of analyzed events; it cannot create new ones. The consequences of analyzed events were evaluated in 1. above. The typographical correction is purely administrative in nature.

3. No. See 1. above. The analysis used to determine the revised SLO MCPR limit meets all pertinent regulatory requirements for use in this application, and concluded that the consequences were but a small fraction of 10 CFR 100 guidelines.

   The typographical correction is purely administrative in nature.

   Based on the above, the proposed changes will not result in a significant decrease in any margin of safety.

Specification 5.3.1, Fuel Assemblies

The proposed changes to this section delete unnecessary references to the initial core loading.

1. No. References to the initial core loading, which has been completely discharged, are unnecessary and proposed to be deleted. The ANF-5 9x9 fuel has similar thermal hydraulic and nuclear operating characteristics to the ANF-4 9x9 design which has been previously approved by the NRC (See Summary Report Reference 7) for coresidence with the ANF 8x8 fuel that will remain in the core. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. As described above, the ANF 9x9 fuel has been previously evaluated for coresidence with ANF 8x8 fuel in the core. No new events have been determined to result from this change.

3. No. Based on its similar operating characteristics, previously approved analyses, and the analyses and limits which are proposed in this application, the U1C6 reload fuel will not result in a significant decrease in any margin of safety.

<u>Specification 5.3.2, Control Rod Assemblies</u>

The changes to this specification are provided in order to recognize the replacement blade design being introduced in U1C6.

1. No. The main differences between the replacement Duralife 160C control blades and the original equipment control blades are:

   a. the Duralife 160C control blades utilize improved B_4C tube material (i.e. high purity stainless steel vs. commercial purity stainless steel) to eliminate cracking during the lifetime of the control blade;

   b. the Duralife 160C control blades utilize three solid hafnium rods at each edge of the cruciform which replace the three B_4C rods that are most susceptible to cracking to increase control blade life;

   c. the Duralife 160C control blades contain additional B_4C tubes in place of the stiffeners, have an increased sheath thickness, utilize a full length weld to attach the handle and velocity limiter, and contain additional coolant holes at the top and bottom of the sheath which result in a crevice-free structure;

   d. the Duralife 160C control blades utilize low cobalt-bearing pin and roller materials in place of stellite which was previously utilized;

   e. the Duralife 160C control blades are longer by approximately 3.1 inches in order to facilitate fuel moves within the reactor vessel during refueling outages at Susquehanna SES; and

   f. the Duralife 160C control blades are approximately 16 pounds heavier as a result of the design changes described above.

   The Duralife control blade has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and any other lateral and vertical loads expected for each condition. The Duralife 160C control blade stresses, strains, and cumulative fatigue have been evaluated and result in an acceptable margin to safety. The control blade insertion capability has been evaluated and it has been determined to be capable of insertion into the core during all modes of plant operation within the limits of plant analyses. The Duralife 160C control blade coupling mechanism is equivalent to the original equipment coupling mechanism and is fully compatible with the existing control rod drives in the plant. In addition, material selected is compatible with the reactor environment. The impact of the increased weight of the control blades on the seismic and hydrodynamic load evaluation of the reactor vessel and internals has been reviewed and found to have a negligible effect on existing analyses.

With the exception of the crevice-free structure and the extended handle, the Duralife 160C control blades are equivalent to the NRC approved Hybrid I Control Blade Assembly (Summary Report Reference 9). The mechanical aspects of the crevice-free structure were approved by the NRC for all control blade designs in Summary Report Reference 10. A neutronics evaluation of the crevice-free structure for the Duralife 160C design was performed by GE using the same methodology as was used for the Hybrid I control blades in Reference 9. These calculations were performed for the original equipment control blades and the Duralife 160C control blades described above assuming an array of ANF 9x9 fuel. The Duralife 160C control blade has a slightly higher worth than the original equipment design, but the increase in worth is within the criterion for nuclear interchangeability. The increase in blade worth has been taken into account in the appropriate U1C6 analyses. However, as stated in Summary Report Reference 9, the current practice in the lattice physics methods is to model the original equipment all B_4C control blade as non-depleted. The effects of control blade depletion on core neutronics during a cycle are small and are inherently taken into account by the generation of a target k-effective for each cycle. As discussed above, the neutronics calculations of the crevice-free structure show that the non-depleted Duralife 160C control blade has direct nuclear interchangeability with the non-depleted original equipment all B_4C design. The Duralife 160C also has the same end-of-life reactivity worth reduction limit as the all B_4C design. Therefore, the Duralife 160C can be used without changing the current lattice physics models as previously approved for the Hybrid I control blades (Summary Report Reference 9).

The extended handle and the crevice-free structure features of the Duralife 160C control blades result in a one pound increase in the control blade weight over that of the Hybrid I blades, and a sixteen pound increase over the Susquehanna SES original equipment control blades. In Summary Report Reference 9, the NRC approved the Hybrid I control blade which weighs less (by more than one pound) than the D lattice control blade. The basis of the Control Rod Drop Accident analysis continues to be conservative with respect to control rod drop speed since the Duralife 160C control blade weighs less than the D lattice control blade, and the heavier D lattice control blade speed is used in the analysis. In addition, GE performed scram time analyses and determined that the Duralife 160C control blade scram times are not significantly different than the original equipment control blade scram times. The current Susquehanna SES measured scram times also have considerable margin to the Technical Specification limits. Since the increase in weight of the Duralife 160C control blades does not significantly increase the measured scram speeds and the safety analyses which involve reactor scrams utilize the Technical Specification limit scram times, the safety analyses are not affected.

Since the Duralife 160C control blades contain solid hafnium rods in locations where the B_4C tubes have failed, and the remaining B_4C rods are manufactured with an improved tubing material (high purity stainless steel vs. commercial purity stainless steel), boron loss due to cracking is not expected. PP&L plans to track the depletion of each control blade and discharge any control blade prior to a ten percent loss in reactivity worth. Therefore, the requirements of IE Bulletin 79-26, Revision 1 do not apply to the Duralife 160C control blades.

Based on the discussion above, the new control blades proposed to be utilized in U1C6 do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The replacement blades can only be evaluated for their effectiveness as part of the overall reactivity control system, which is evaluated in terms of analytical consequences in 1. above. Since they do not cause any significant change in system operation or function, no new events are created.

3. No. The analyses described in 1. above indicate that the replacement blades meet all pertinent regulatory criteria for use in this application, and are expected to eliminate the boron loss concerns expressed in IE Bulletin 79-26, Revision 1. Therefore, the proposed change does not result in a significant decrease in any margin of safety.

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