ATTACHMENT 3

TECHNICAL SPECIFICATION CHANGES

DEFINITIONS

SECTION

Replace with,

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Delete

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Delete

Delete

ADMINISTRATIVE CONTROLS

Add

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, LPRMs, TIPs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be the highest value of the FLPD which exists in the core.

Delete and replace with Insert 1

CRITICAL POWER RATIO

1.9 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of a General Electric critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DRYWELL INTEGRITY

1.11 DRYWELL INTEGRITY shall exist when:

a. All drywell penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position.

b. The drywell equipment hatch is closed and sealed.

c. The drywell head is installed and sealed.

d. The drywell air lock is in compliance with the requirements of Specification 3.6.2.3.

e. The drywell leakage rates are within the limits of Specification 3.6.2.2.

Insert 1

CORE OPERATING LIMITS REPORT

1.8 The CORE OPERATING LIMITS REPORT is the Perry Unit 1-specific document that provides the core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these operating limits is addressed in individual Specifications.

POWER DISTRIBUTION LIMITS

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed the result obtained from multiplying the applicable MAPLHGR values* by the smaller of either the flow dependent MAPLHGR factor ($MAPFAC_f$) of Figure 3.2.1-1 or the power dependent MAPLHGR factor ($MAPFAC_p$) of Figure 3.2.1-2 limits specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

If at any time during operation it is determined that an APLHGR is exceeding limits specified in the CORE OPERATING LIMITS REPORT, the result of the above multiplication, 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 above limits:

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 in one hour, 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.

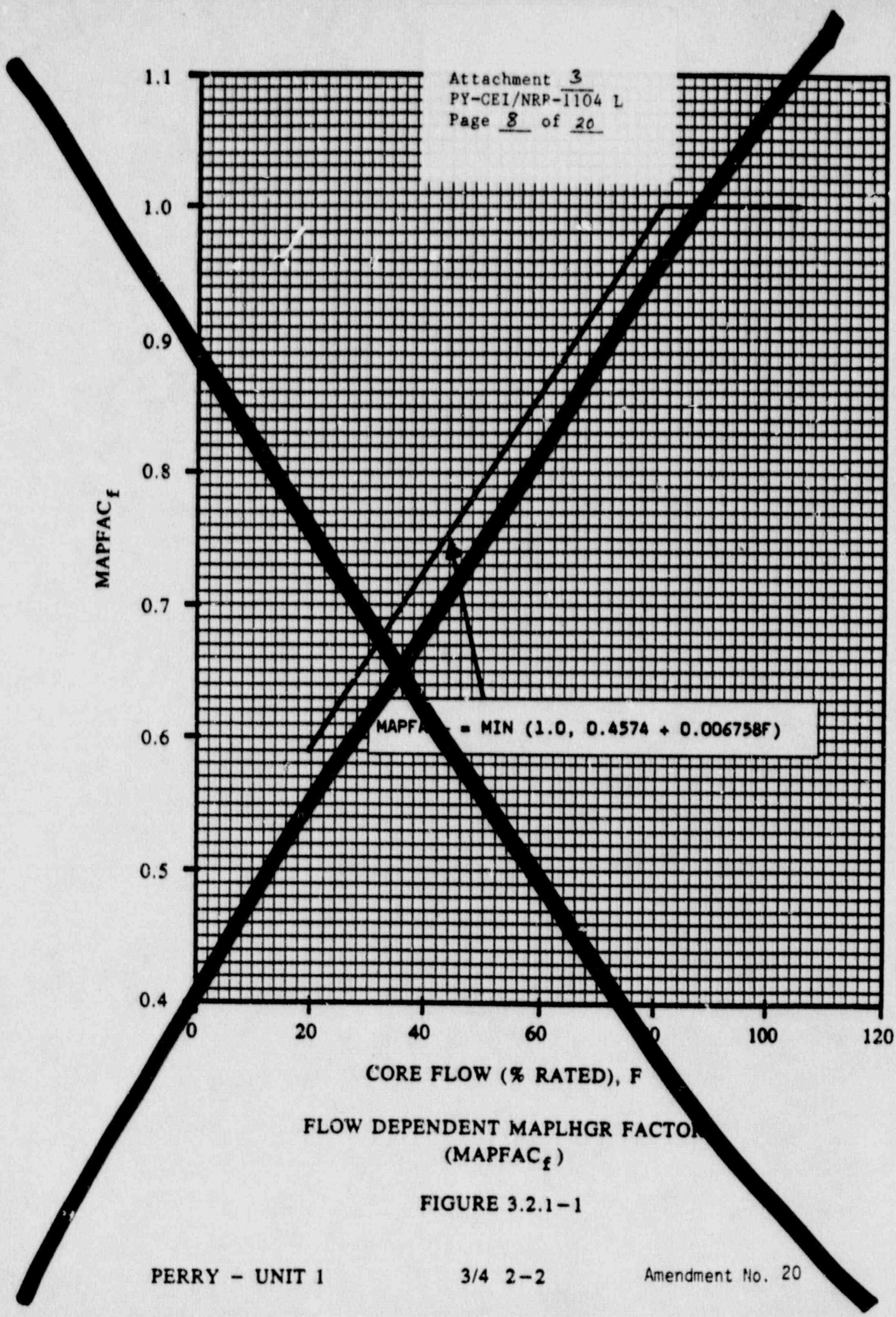
---

* These applicable MAPLHGR values are:

1) Those that have been approved for the respective fuel and lattice type as a function of the average planar exposure (as determined by the NRC approved methodology described in GESTAR-II)

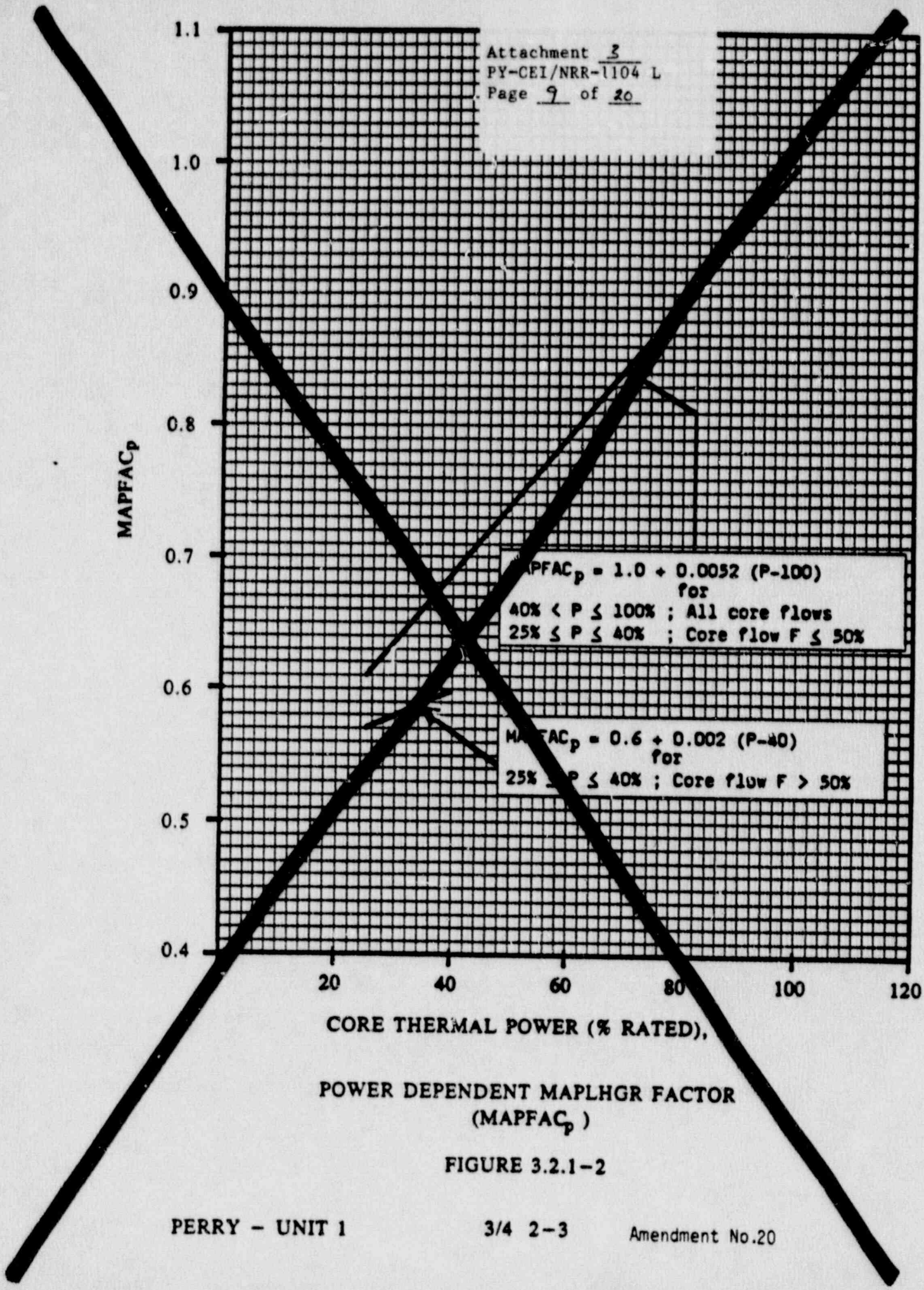
or

2) When hand calculations are required, the MAPLHGR as a function of the average planar exposure for the most limiting lattice (excluding natural uranium) shown in the Figures 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6 for the applicable type of fuel.



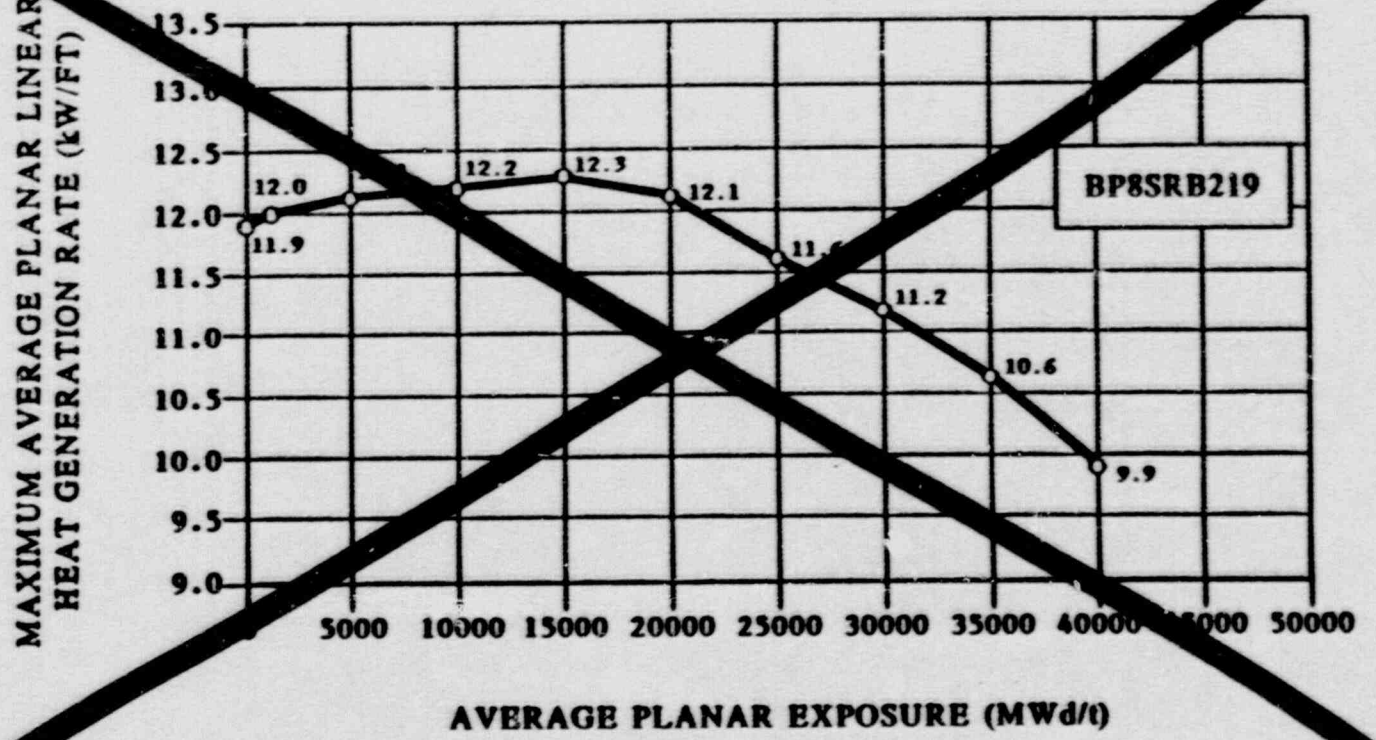
FLOW DEPENDENT MAPLHGR FACTOR
(MAPFAC$_f$)

FIGURE 3.2.1-1



POWER DEPENDENT MAPLHGR FACTOR
(MAPFAC$_p$)

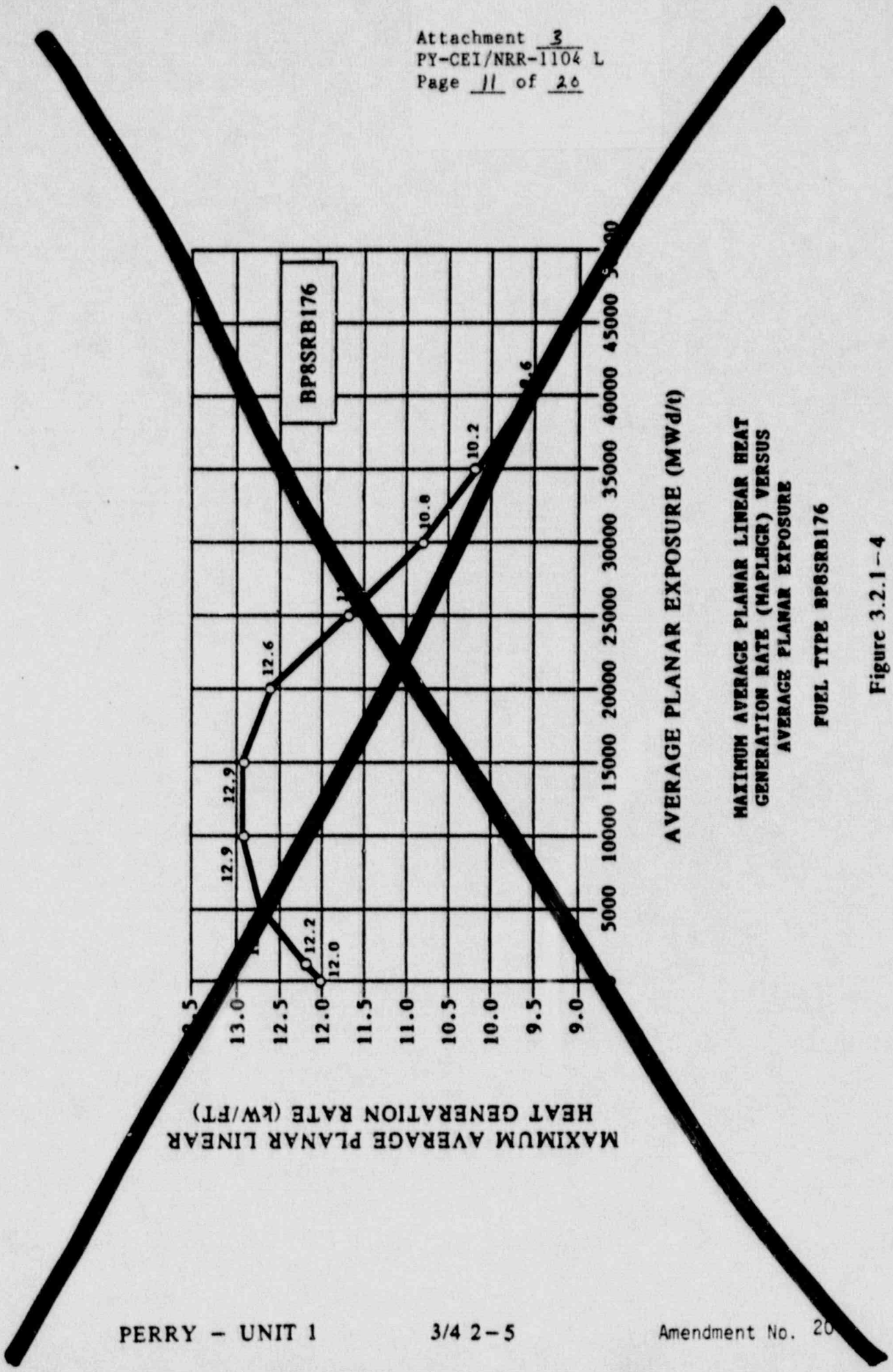
FIGURE 3.2.1-2



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FUEL TYPE BP8SRB219

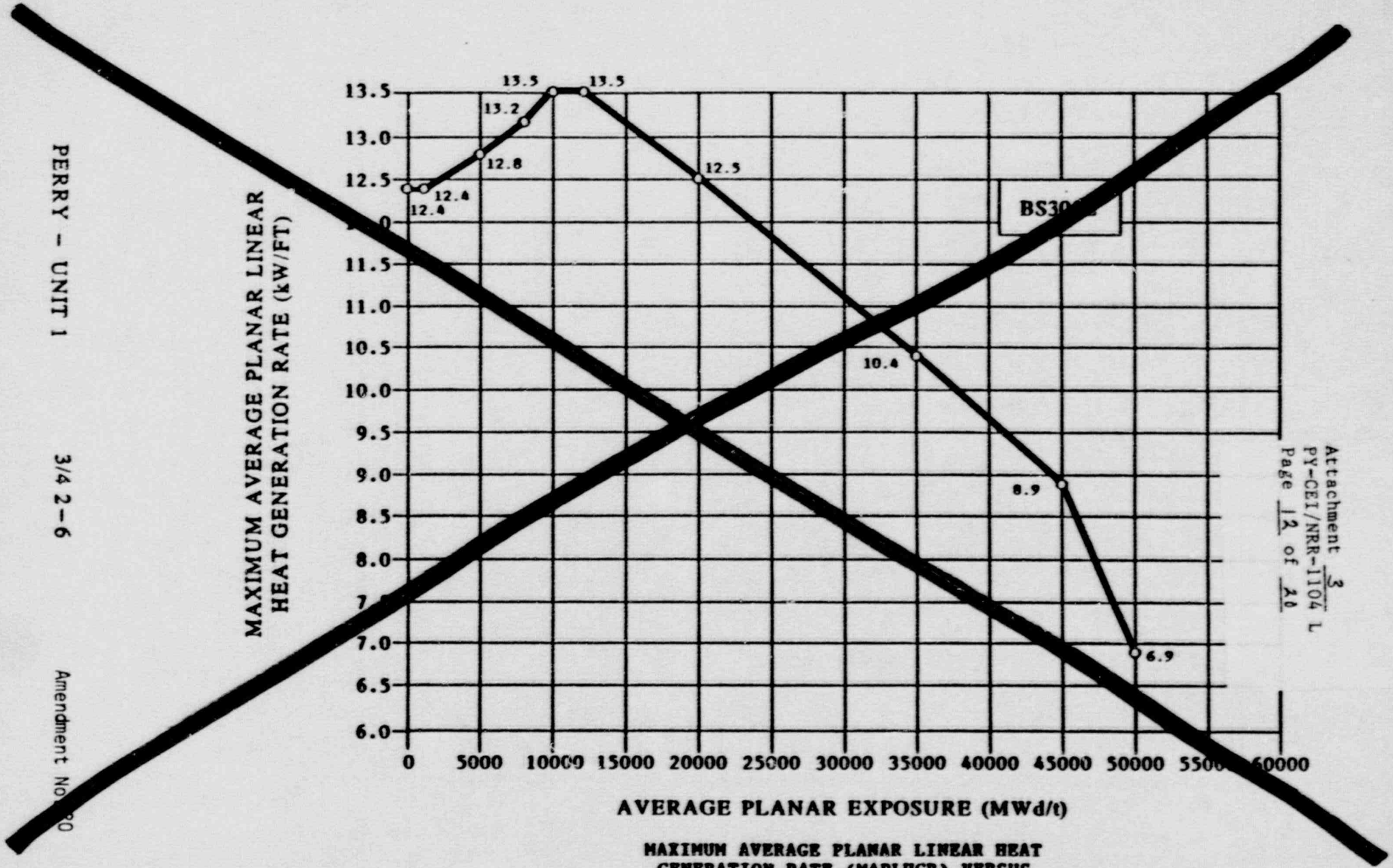
Figure 3.2.1–3



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FUEL TYPE BP8SRB176

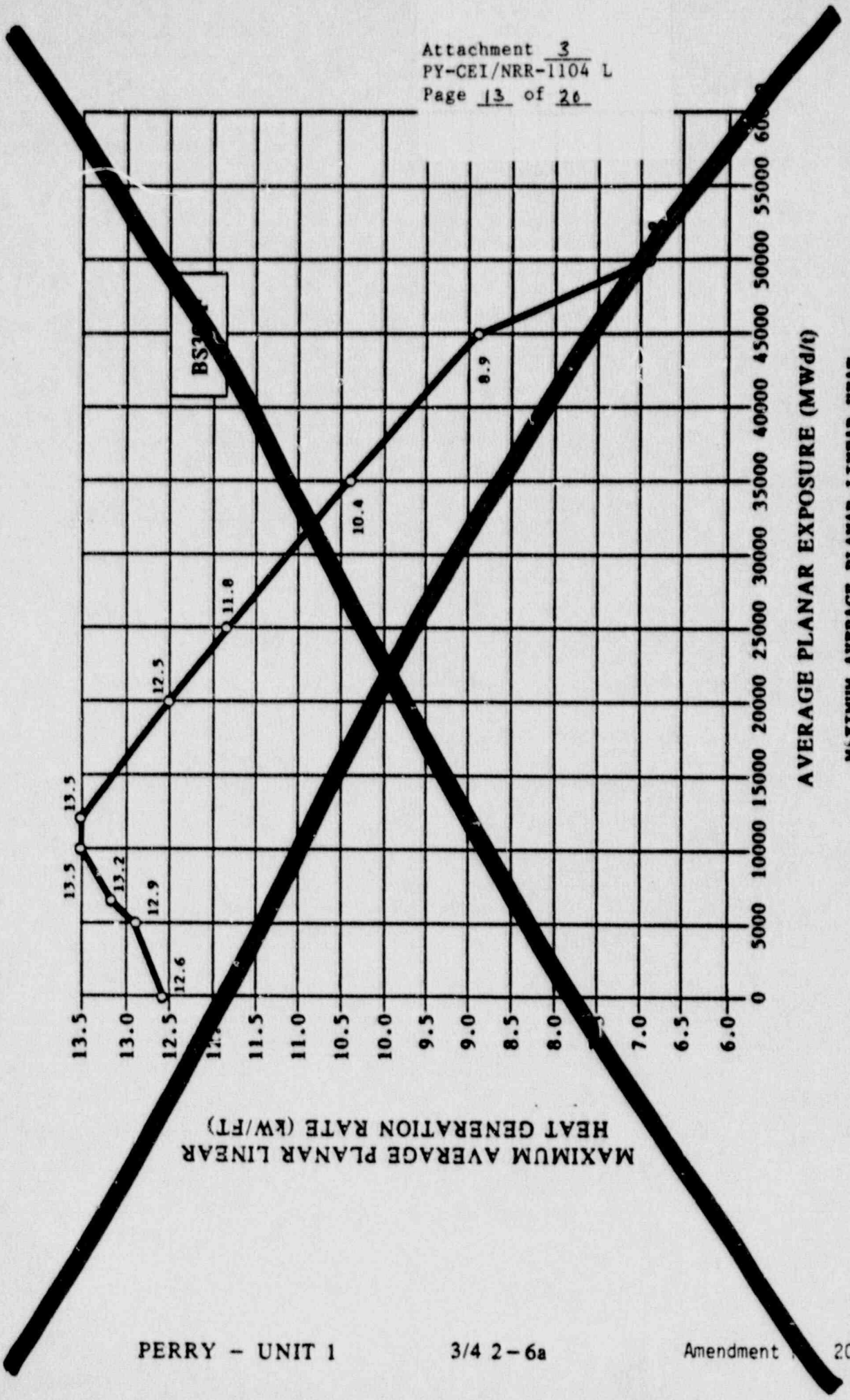
Figure 3.2.1-4



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, GE8x8EB

FUEL TYPE BS301E

Figure 3.2.1–5



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, GE8x8EB

FUEL TYPE BS301P

Figure 3.2.1-6

POWER DISTRIBUTION LIMITS

3/4.2.2 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than both the MCPR$_f$ and MCPR$_p$ limits specified in the CORE OPERATING LIMITS REPORT at the indicated core flow, THERMAL POWER, ΔT* and core average exposure compared to End of Cycle Exposure (EOCE)** as shown in Figures 3.2.2-1 and 3.2.2-2.

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

ACTION:

With MCPR less than the applicable MCPR limit shown in Figures 3.2.2-1 and 3.2.2-2 limits specified in the CORE OPERATING LIMITS REPORT, 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

4.2.2 MCPR shall be determined to be equal to or greater than the MCPR$_{SA}$ limit determined from Figures 3.2.2-1 and 3.2.2-2 specified in the CORE OPERATING LIMITS REPORT:

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 in one hour, and

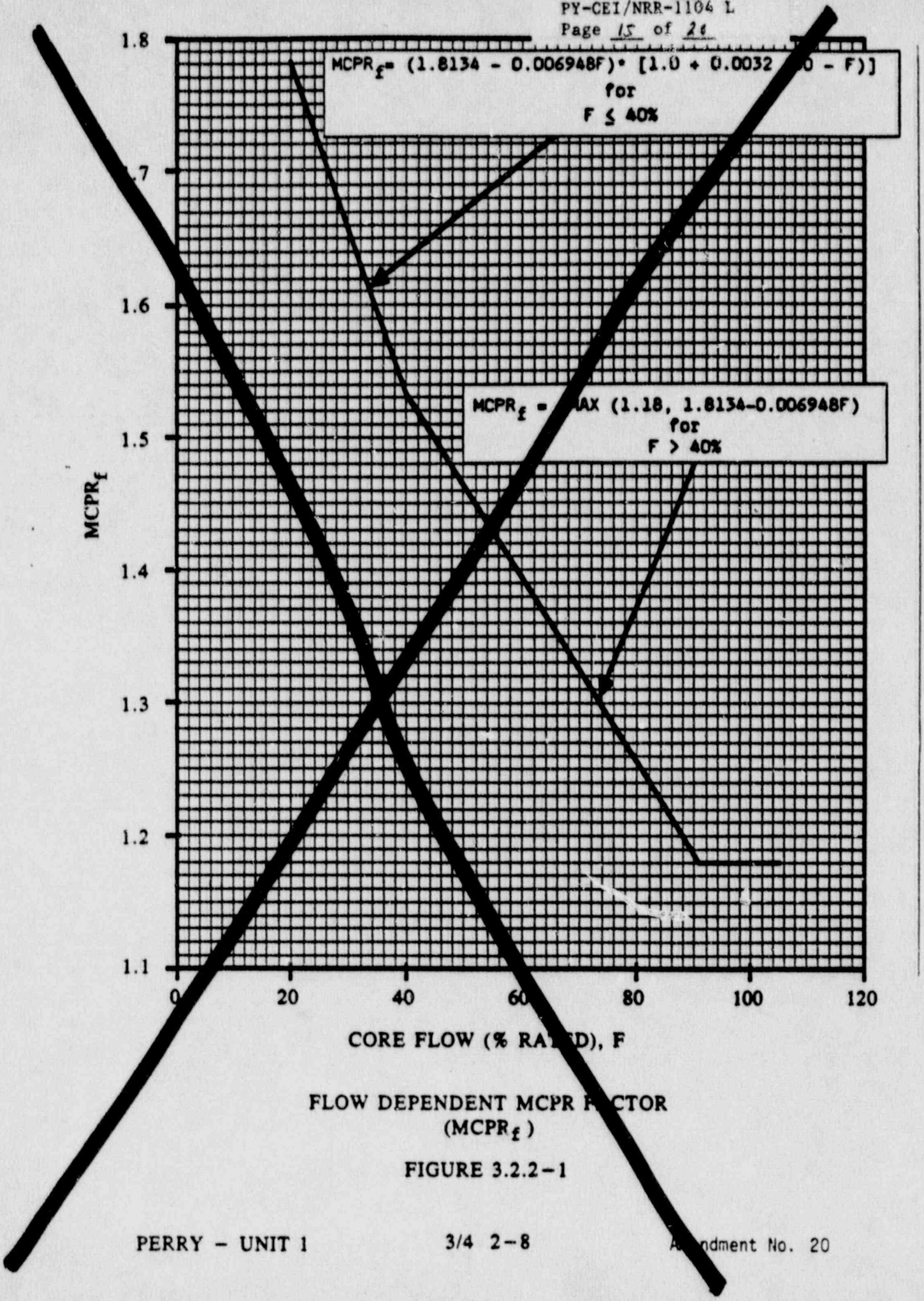
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

---

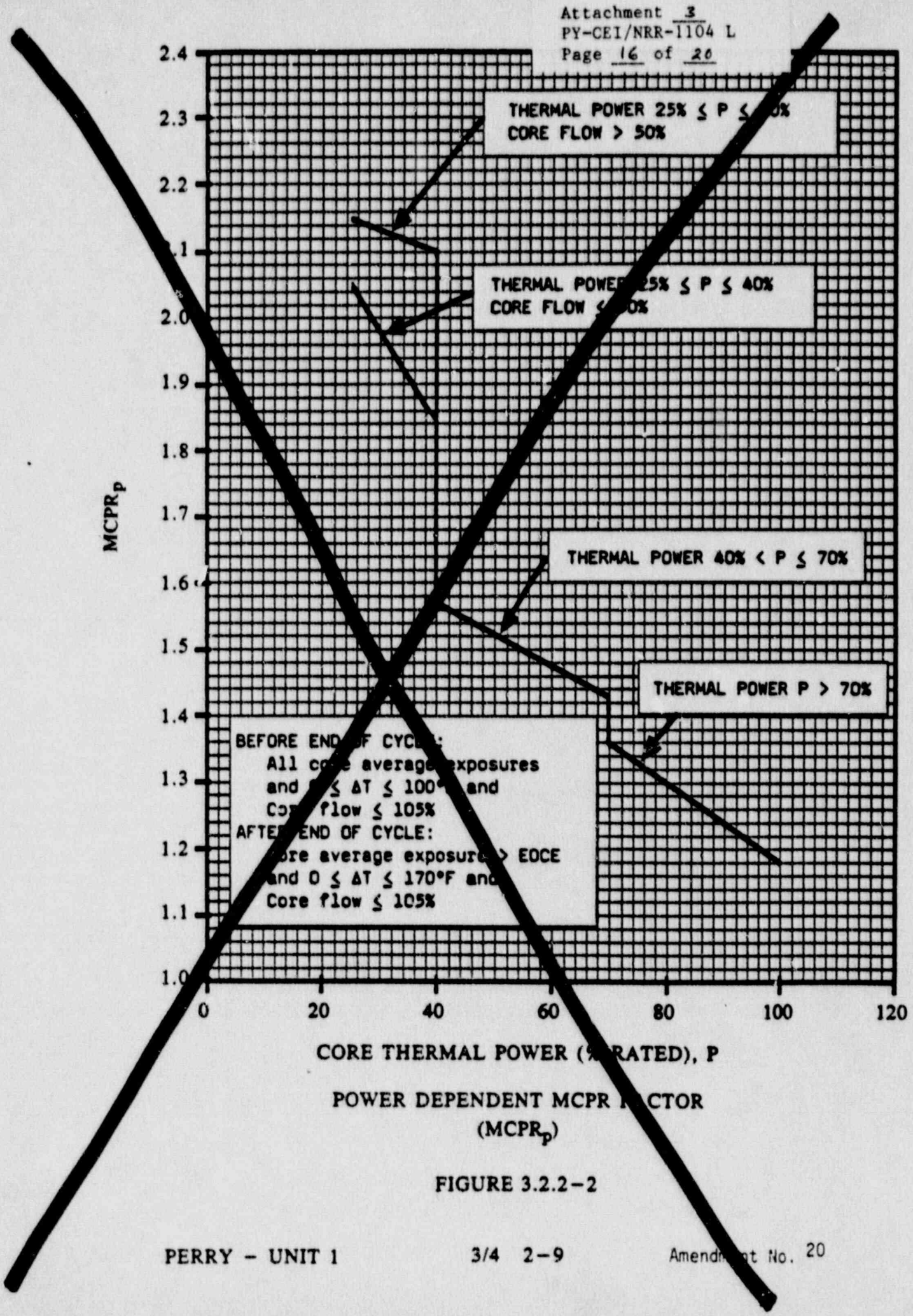
*This ΔT refers to the planned reduction of rated feedwater temperature from nominal rated feedwater temperature (420°F), such as prolonged removal of feedwater heater(s) from service.

**End of Cycle Exposure (EOCE) is defined as 1) the core average exposures at which there is no longer sufficient reactivity to achieve RATED THERMAL POWER with rated core flow, all control rods withdrawn, all feedwater heaters in service and equilibrium Xenon, or 2) as specified by the fuel vendor.



FLOW DEPENDENT MCPR FACTOR
(MCPR$_f$)

FIGURE 3.2.2-1



POWER DEPENDENT MCPR FACTOR
($MCPR_p$)

FIGURE 3.2.2-2

POWER DISTRIBUTION LIMIT

3/4.2.3 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.3 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limit specified in the CORE OPERATING LIMITS REPORT.

a. 13.4 kw/ft for BP8x8R fuel.

b. 14.4 kw/ft for GE8x8EB fuel.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 LHGR's 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 in one hour, 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.

DESIGN TEMPERATURE AND PRESSURE (Continued)

b. Maximum internal temperature:
   1. Drywell 330°F.
   2. Suppression pool 185°F.

c. Maximum external to internal differential pressure:
   1. Drywell 21 psid.
   2. Containment 0.8 psid.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the annulus between the shield building and the primary containment and has a minimum free volume of 392,548 cubic feet.

5.3 REACTOR CORE

and shall be limited to those fuel assemblies which have been analyzed with NRC-approved codes and methods, ~~and~~ have been shown to comply with all Safety Design Bases in the FSAR, and are identified in the CORE OPERATING LIMITS REPORT.

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 748 fuel assemblies with each fuel assembly containing 62 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 1.9 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 177 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143.7 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 pump.

ADMINISTRATIVE CONTROLS

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.7.9 or 3.3.7.10, respectively; and description of the events leading to liquid holdup tanks exceeding the limits of Specification 3.11.1.4.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office no later than the 15th of each month following the calendar month covered by the report.

SPECIAL REPORTS Insert 2

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office within the time period specified for each report.

6.9.3 Safety/relief valve failures will be reported to the Regional Administrator of the Regional Office of the NRC via the License Event Report system within 30 days.

6.9.4 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be reported to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report system within 30 days.

6.10 RECORD RETENTION

Move to new Page 6-21a

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

Insert 2

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in the General Electric reload licensing topical report; General Electric Standard Application for Reactor Fuel - GESTAR II: NEDE-24011-P-A and US (latest approved revision). The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

ATTACHMENT 4

TECHNICAL SPECIFICATION

BASES CHANGES

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 (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10 CFR 50.46 and that the fuel design analysis limits specified in GESTAR-II (Reference 1) will not be exceeded.

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 peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The MAPLHGR limits of Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 are multiplied by the smaller of either the flow dependent MAPLHGR factor ($MAPFAC_f$) or the power dependent MAPLHGR factor ($MAPFAC_p$) corresponding to existing core flow and power state to assure the adherence to fuel mechanical design bases during the most limiting transient. $MAPFAC_f$'s are determined using the three-dimensional BWR simulator code to analyze slow flow runout transients. $MAPFAC_p$'s are generated using the same data base as the $MCPR_p$ to protect the core from plant transients other than core flow increases.

specified in the CORE OPERATING LIMITS REPORT

The Technical Specification MAPLHGR value is the most limiting composite of the fuel mechanical design analysis MAPLHGR and the ECCS MAPLHGR.

Fuel Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice, operating at the bounding power history, meet the fuel design limits specified in Reference 1. This bounding power history is used as the basis for the fuel design analysis MAPLHGR value.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR Part 50 Appendix K to demonstrate that the MAPLHGR values comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

POWER DISTRIBUTION LIMITS

BASES

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

Only the most limiting MAPLHGR values are shown in the Technical Specification figures for multiple lattice fuel. When hand calculations are required, these Technical Specification MAPLHGR figure values for that fuel type are used for all lattices in that bundle.

For some GE fuel bundle designs MAPLHGR depends only on bundle type and burnup. Other GE fuel bundles have MAPLHGRs that vary axially depending upon the specific combination of enriched uranium and gadolinia that comprises a fuel bundle cross section at a particular axial node. Each particular combination of enriched uranium and gadolinia, for these fuel bundle types, is called a lattice type by GE. These particular fuel bundle types have MAPLHGRs that vary by lattice type (axially) as well as with fuel burnup.

Approved MAPLHGR values (limiting values of APLHGR) as a function of fuel and lattice types, and as a function of the average planar exposure are provided in Technical Specification Figures 3.2.1-3 through 3.2.1-6 the CORE OPERATING LIMITS REPORT.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.2 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.2 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07 and an analysis of the limiting 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 are documented in the USAR and Reference 1. The limiting transient yields the largest delta CPR. When added to the Safety Limit MCPR, the required operating limit MCPR of Specification 3.2.2 is obtained. The power-flow map of Figure B 3/4 2.2-1 defines the analytical basis for generation of the MCPR operating limits.

The evaluation of a given transient begins with the system initial parameters shown in USAR Chapter 15 and/or Reference 1, and Cleveland Electric's November 28 and December 29, 1988 submittals that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate these events are described in Reference 1.

The purpose of the $MCPR_f$ and $MCPR_p$ is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_p$ figure contained in the CORE OPERATING LIMITS REPORT Figure 3.2.2-2 also reflects the required MCPR values resulting from the analysis performed to justify operation with the feedwater temperature ranging from 420°F to 320°F at 100% RATED THERMAL POWER steady state conditions, and also beyond the end of cycle with the feedwater temperature ranging from 420°F and 250°F.

The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along a conservative steep generic power flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along this conservative steep power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

ATTACHMENT 5

CORE OPERATING LIMITS REPORT

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

PERRY NUCLEAR POWER PLANT OPERATIONS MANUAL

Plant Data Book - TAB F

TITLE: UNIT 1 CORE OPERATING LIMITS REPORT
CYCLE 2 (RELOAD 1)

REVISION : 0 **EFFECTIVE DATE:** _____

PREPARER: _____ / Date

REVIEWER: _____ / Date

PORC MEETING NO.: _____ / Date

APPROVED: _____ / Date

CORE OPERATING LIMITS REPORT
PERRY NUCLEAR POWER PLANT - UNIT 1
CYCLE 2 (RELOAD 1) REVISION 0

TABLE OF CONTENTS

TABLE OF CONTENTS

NOTE: The preparer and reviewer and responsible for verifying that all Plant Data Book entries shown in the above Table of Contents are correct and reflect the fuel/core design information. Individual entries for this section of the Plant Data Book, Tab F - Core Operating Limits Report, may not be updated without a corresponding revision to this entry, PDB-F-001 and subsequent PORC review of this and any other revised entries. Perry Plant Technical Department Director's approval is required for issuance of any revisions to the Core Operating Limits Report.

SCOPE OF REVISION

Affected Revision	Affected PDB Entry(s)	Summary of Changes	Date
0	All	Original Issue	3/15/90

INTRODUCTION AND REFERENCES

INTRODUCTION

This Core Operating Limits Report for PNPP Unit 1 Cycle 2 is prepared in accordance with the requirements of PNPP Technical Specification 6.9.1.9. The core operating limits presented here were developed using NRC-approved methods (Reference 2). Results from the reload analyses for the General Electric fuel in PNPP Unit 1 Cycle 2 are documented in References 3 and 4.

The cycle-specific core operating limits for the following PNPP Unit 1 Technical Specifications are included in this report:

1. Average Planar Linear Heat Generation Rate (APLHGR) Limit (Technical Specification 3/4.2.1)

2. Minimum Critical Power Ratio Operating Limit (Technical Specification 3/4.2.3)

3. Linear Heat Generation Rate (LHGR) Limit (Technical Specification 3/4.2.4)

REFERENCES

1. USNRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988.

2. "General Electric Standard Application for Reactor Fuel-GESTAR II," NEDE-24011-P-A (latest approved revision) and NEDE-24011-P-A-US (US Supplement - latest approved revision).

3. "Supplemental Reload Licensing Submittal for the Perry Nuclear Power Plant Unit 1, Reload 1, Cycle 2," GE Document 23A5948 Rev. 1. (November 1988).

4. "Supplement 1 to the Supplemental Reload Licensing Submittal for the Perry Nuclear Power Plant Unit 1, Reload 1, Cycle 2," GE Document 23A5948AA Rev. 0. (October 1988).

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (TS 3.2.1)

All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed the result obtained from multiplying the applicable MAPLHGR values * by the smaller of either the flow dependent MAPLHGR factor ($MAPFAC_f$) entry PDB-F-101, or the power dependent MAPLHGR factor ($MAPFAC_p$) entry PDB-F-102.

---

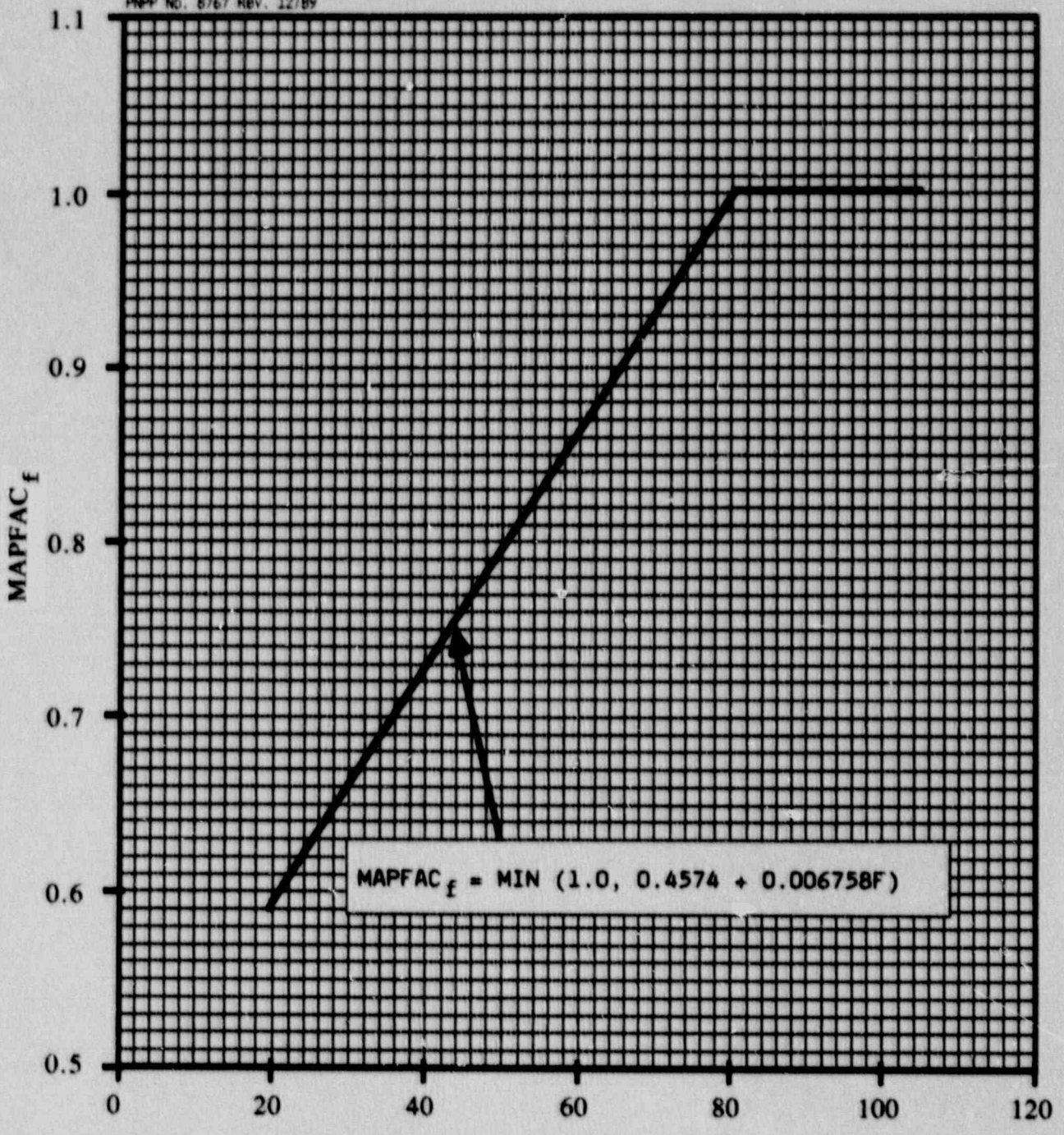
* These applicable MAPLHGR values are:

1. Those that have been approved for the respective fuel and lattice type as a function of the average planar exposure (as described by the NRC approved methodology described in GESTAR-II)

or,

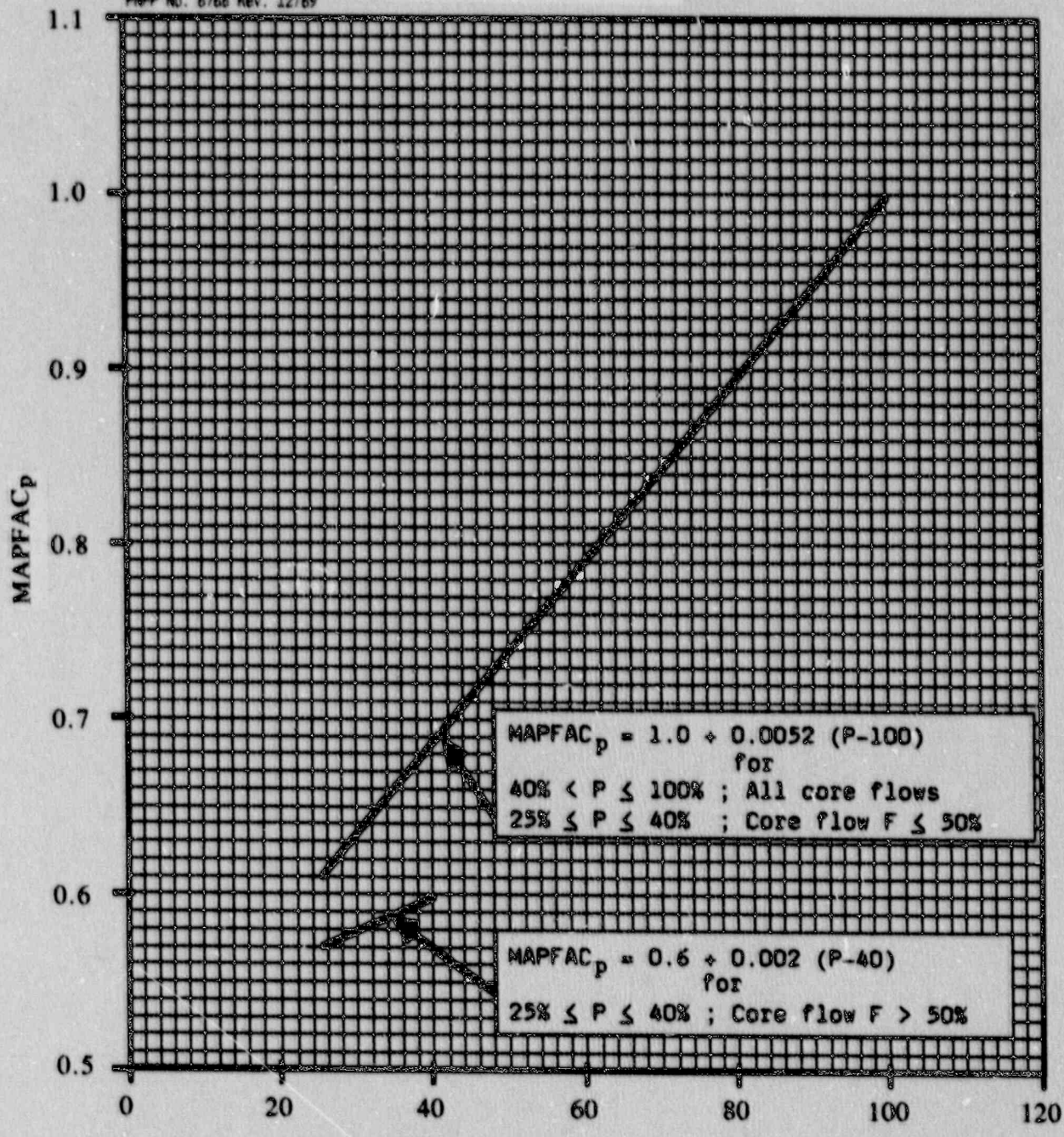
2. When hand calculations are required, the MAPLHGR as a function of the average planar exposure for the most limiting lattice (excluding natural uranium) shown in entries PDB-F-103 through PDB-F-199 for the applicable type of fuel.

PNPP No. 8767 Rev. 12/89



FLOW DEPENDENT MAPLHGR FACTOR
(MAPFAC$_f$)

PNPP No. 8768 Rev. 12/89

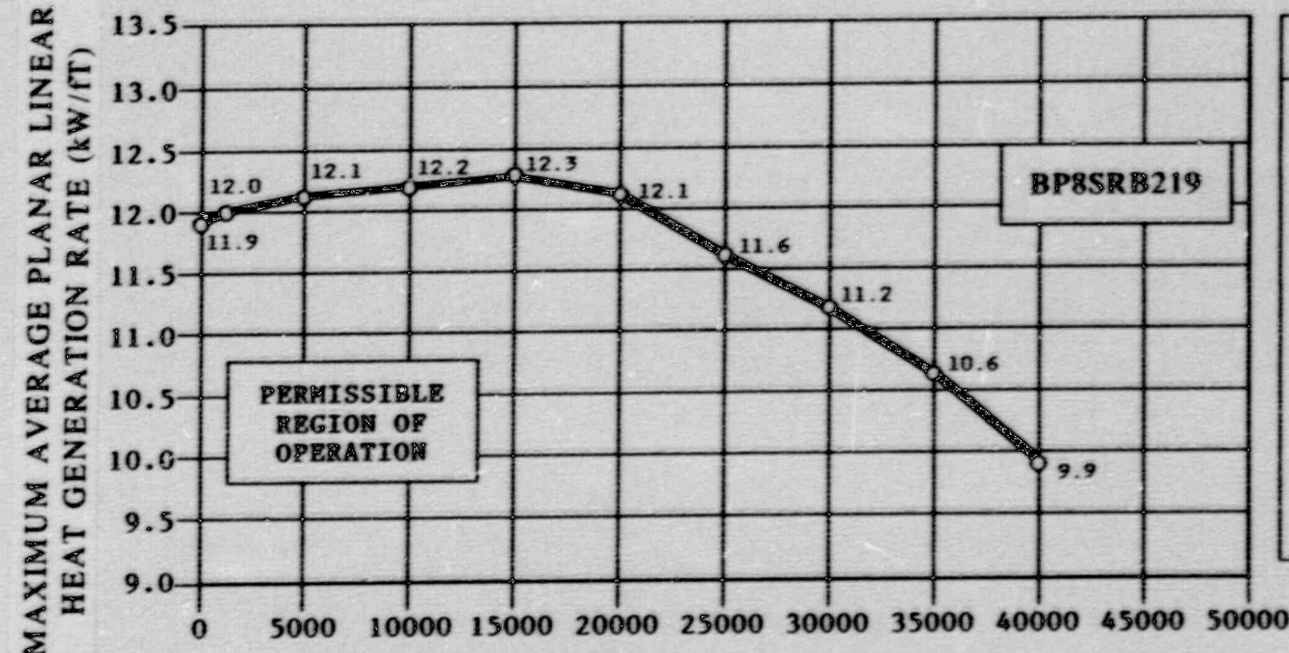


CORE THERMAL POWER (% RATED), P

POWER DEPENDENT MAPLHGR FACTOR

(MAPFAC$_p$)

PRPP No. 8769



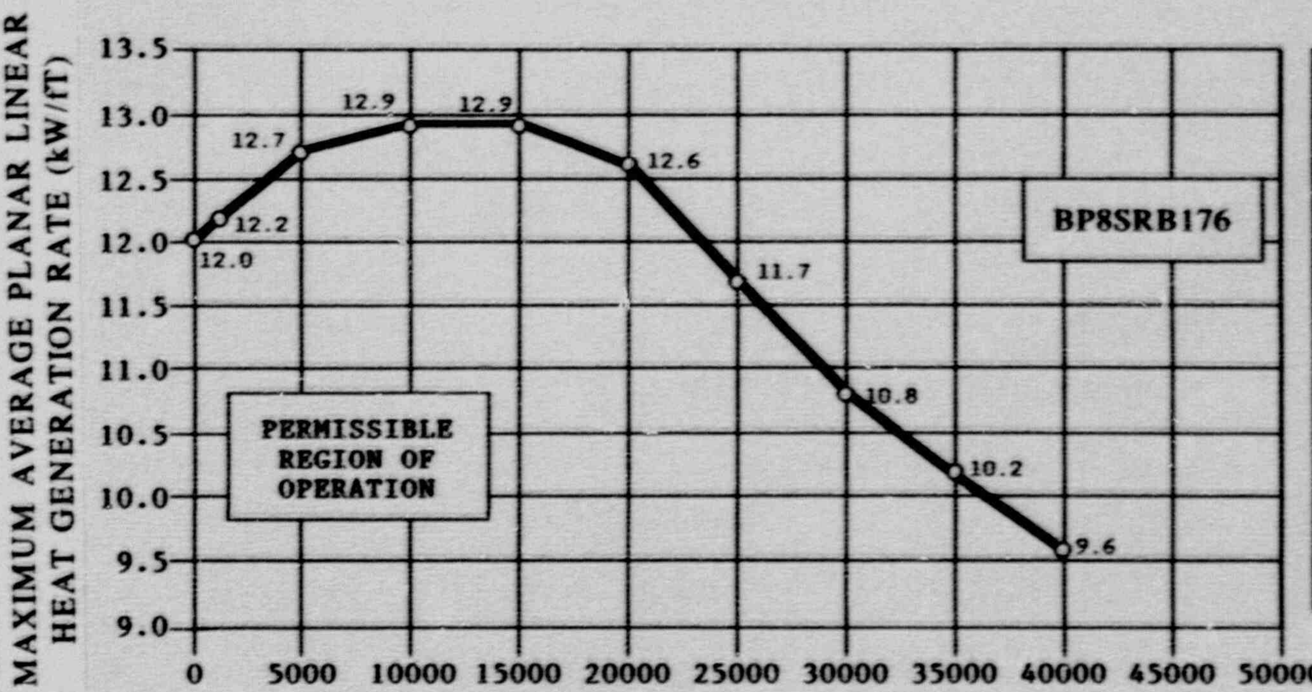
EXPOSURE (MWd/t)	MAPLHGR (kW/ft)
0.0	11.9
1.0	12.0
5.0	12.1
10.0	12.2
15.0	12.3
20.0	12.1
25.0	11.6
30.0	11.2
35.0	10.6
40.0	9.9

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, BP8x8R

FUEL TYPE BP8SRB219

Note: Intermediate MAPLHGR values are obtained by linear interpolation between adjacent points.

PNPP No. 8770

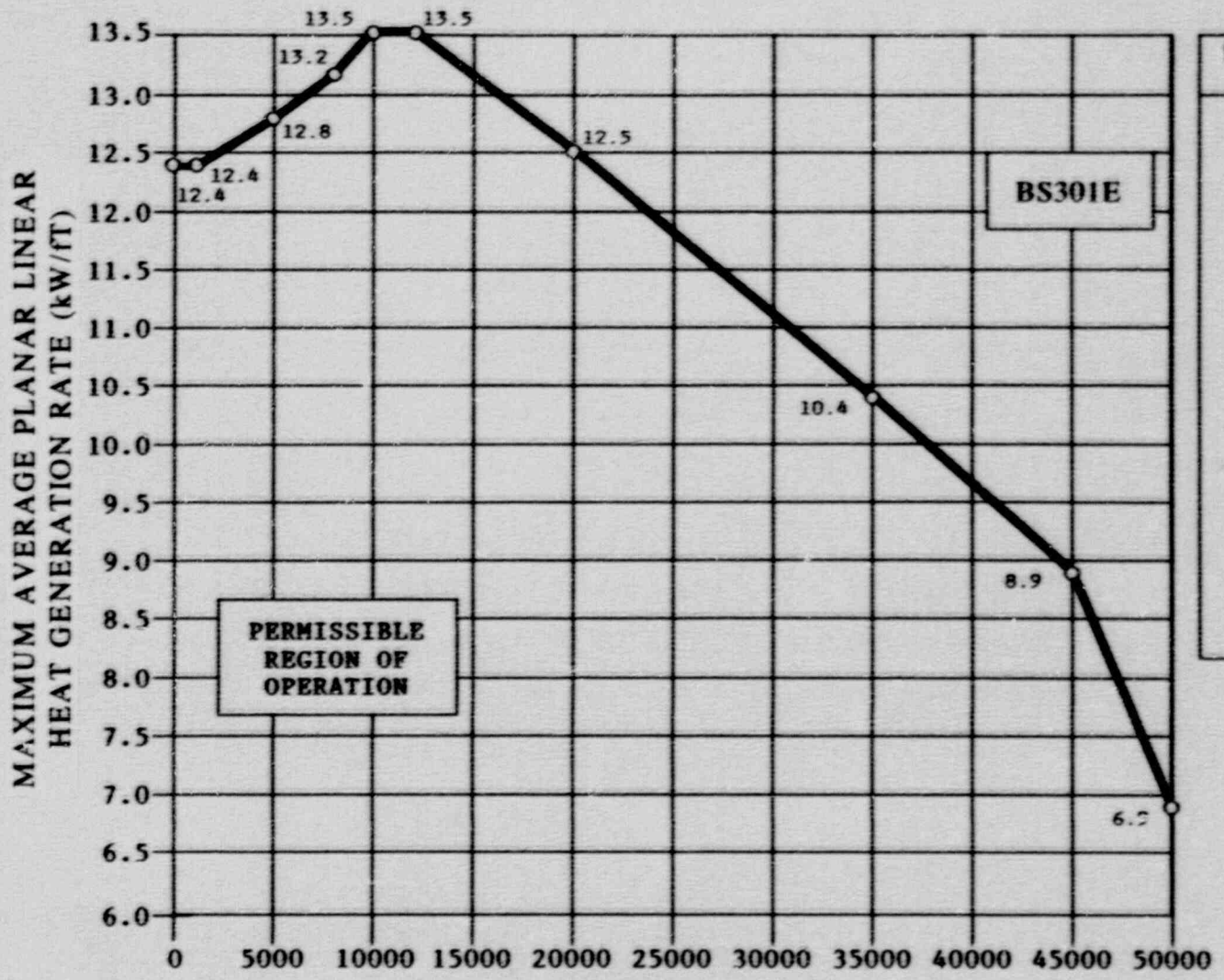


EXPOSURE (MWd/t)	MAPLHGR (kW/ft)
0.0	12.0
1.0	12.2
5.0	12.7
10.0	12.9
15.0	12.9
20.0	12.6
25.0	11.7
30.0	10.8
35.0	10.2
40.0	9.6

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, BP8x8R

FUEL TYPE BP8SRB176

Note: Intermediate MAPLHGR values are obtained by linear interpolation between adjacent points.



PNPP No. 8771

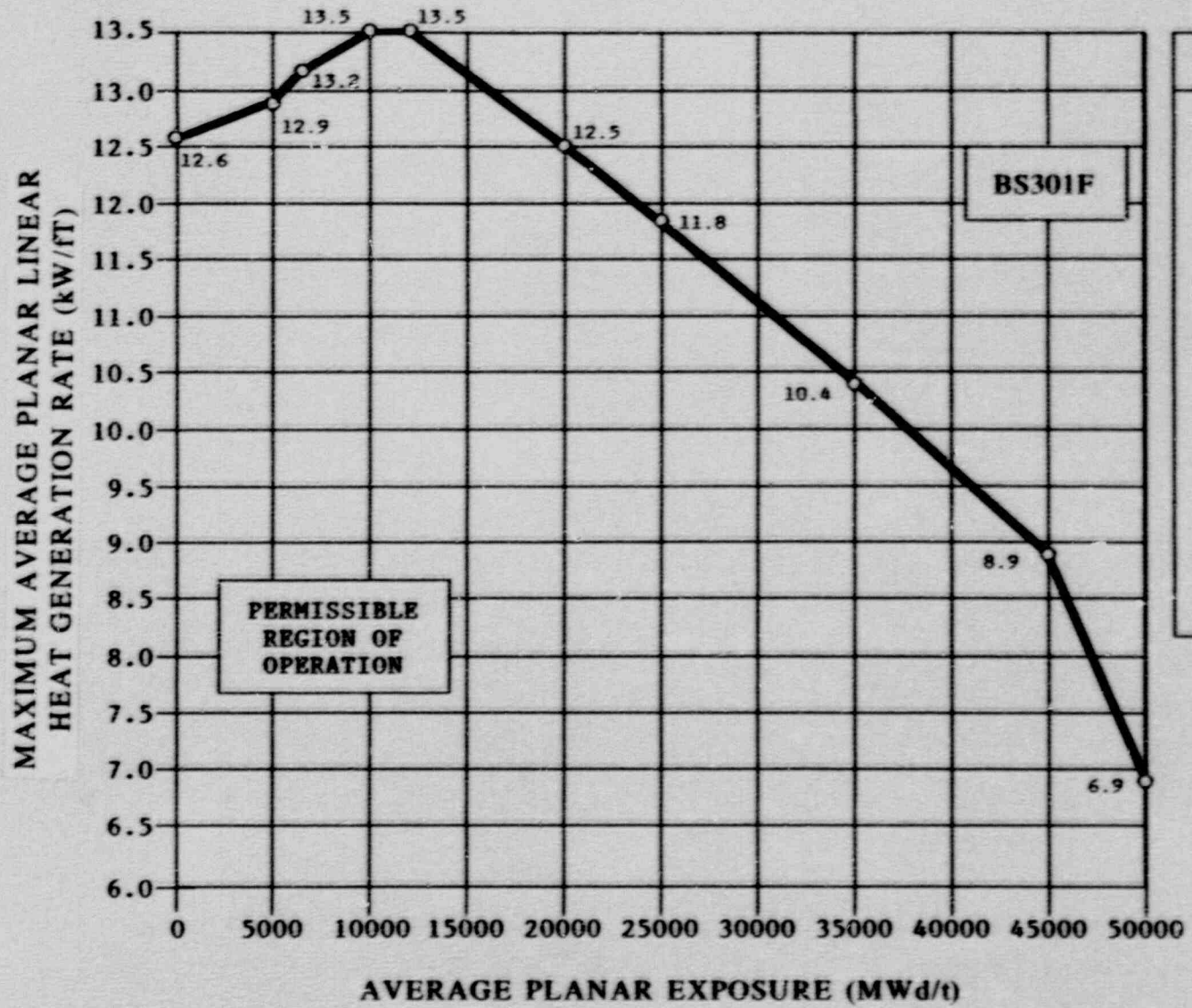
EXPOSURE (MWd/t)	MAPLHGR (kW/ft)
0.0	12.4
1.0	12.4
5.0	12.8
7.0	-
8.0	13.2
10.0	13.5
12.5	13.5
20.0	12.5
25.0	-
35.0	10.4
45.0	8.9
50.0	6.9

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, GE8x8EB

FUEL TYPE BS301E

Notes: 1. Intermediate MAPLHGR values are obtained by linear interpolation between adjacent points.

2. This curve is a composite of the most limiting enriched fuel lattices. For lattice specific values consult Supplement 1 to the Supplemental Reload Licensing Submittal.



EXPOSURE (MWd/t)	MAPLHGR (kW/ft)
0.0	12.6
1.0	-
5.0	12.9
7.0	13.2
8.0	-
10.0	13.5
12.5	13.5
20.0	12.5
25.0	11.8
35.0	10.4
45.0	8.9
50.0	6.9

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, GE8x8EB

FUEL TYPE BS301F

Notes:
1. Intermediate MAPLHGR values are obtained by linear interpolation between adjacent points.
2. This curve is a composite of the most limiting enriched fuel lattices. For lattice specific values consult Supplement 1 to the Supplemental Reload Licensing Submittal.

MINIMUM CRITICAL POWER RATIO (TS 3.2.2)

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than both the $MCPR_f$ and $MCPR_p$ limits at the indicated core flow, THERMAL POWER, delta T * and core average exposure compared to the End of Cycle Exposure (EOCE)** as specified in entries PDB-F-201 and PDB-F-202.

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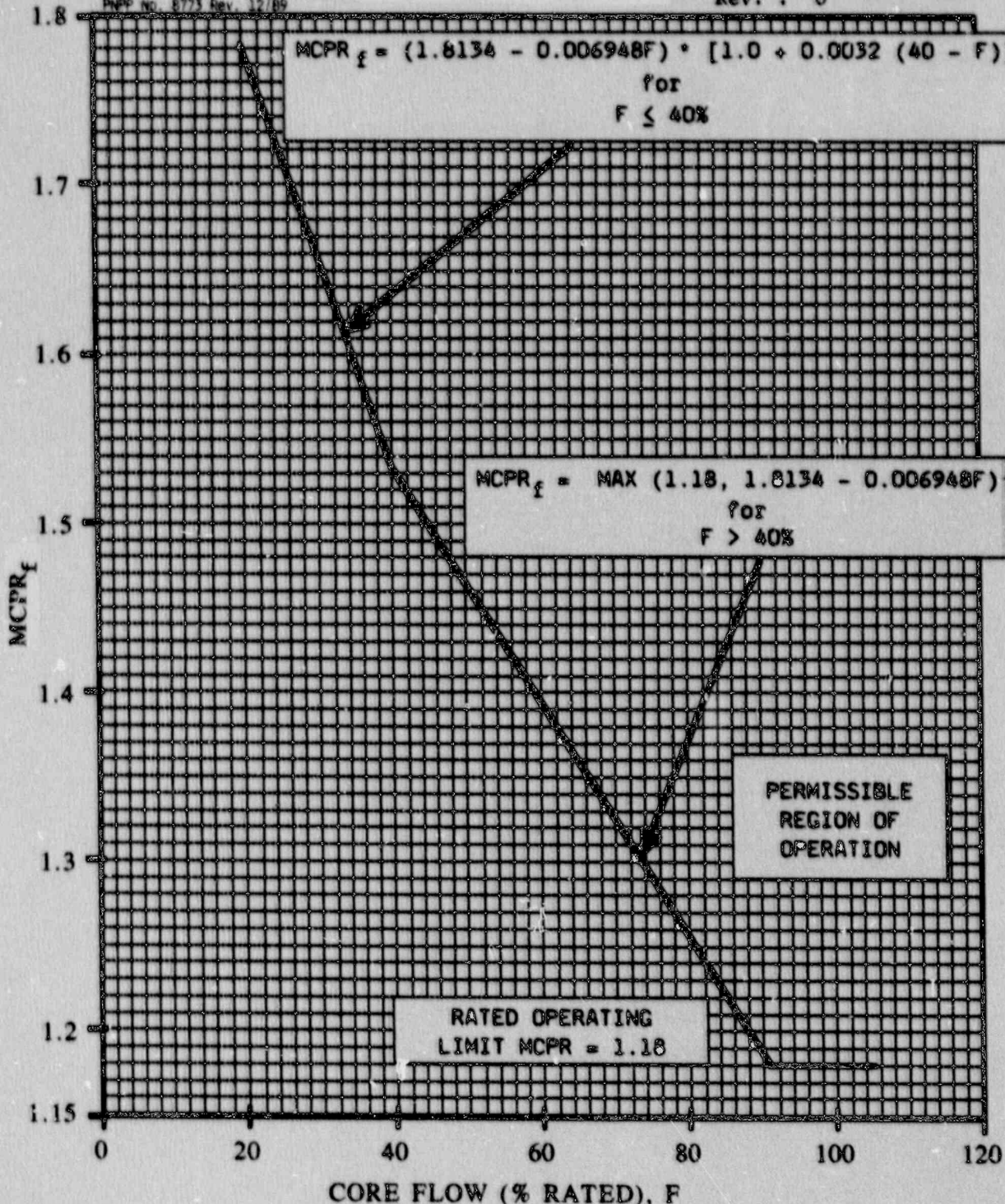
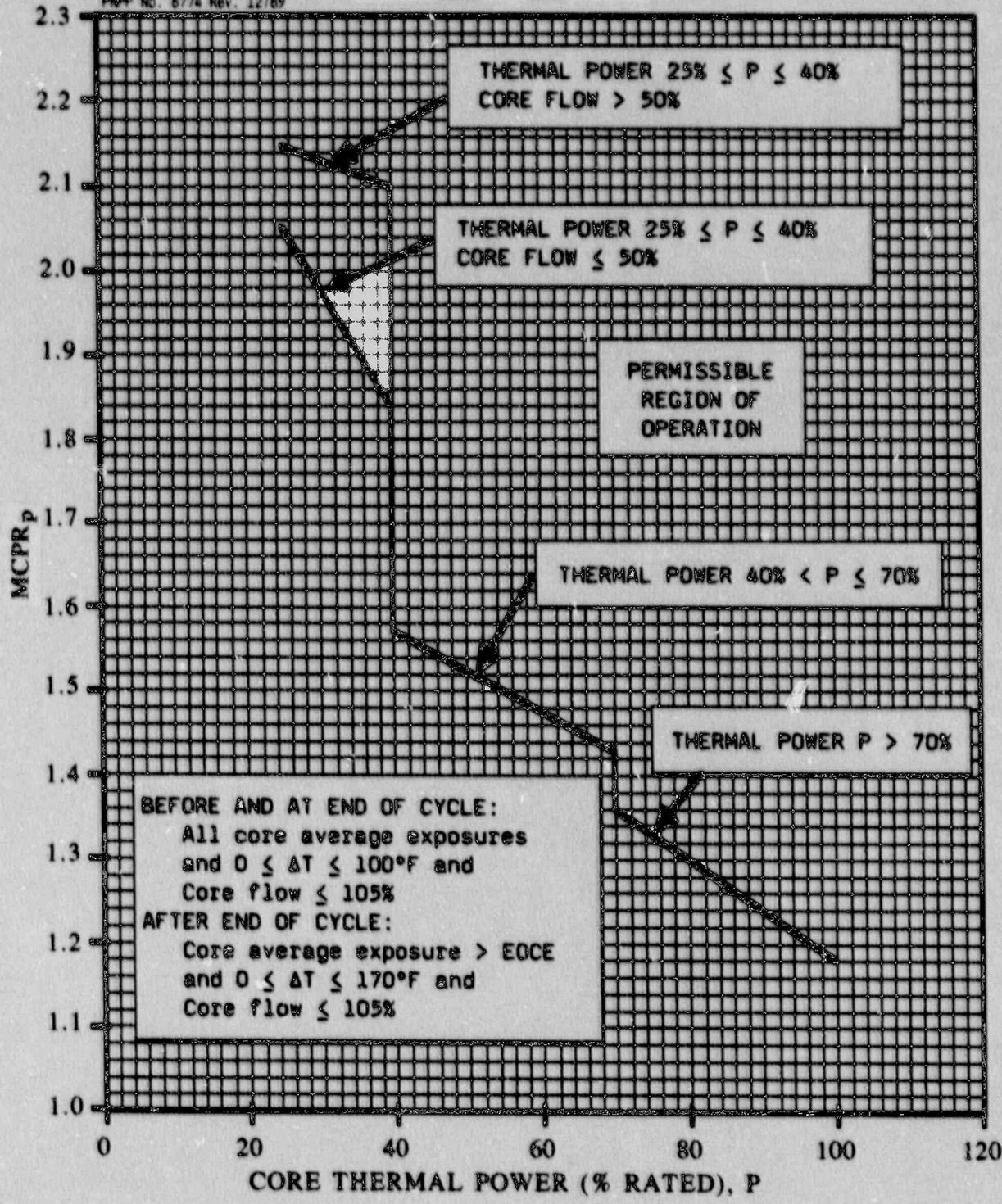
* This delta T refers to the planned reduction of rated feedwater temperature from nominal rated feedwater temperature (420°F), such as prolonged removal of feedwater heater(s) from service.

** End of Cycle Exposure (EOCE) is defined as 1) the core average exposures at which there is no longer sufficient reactivity to achieve RATED THERMAL POWER with rated core flow, all control rods withdrawn, all feedwater heaters in service and equilibrium Xenon, or 2) as specified by the fuel vendor.



FLOW DEPENDENT MCPR FACTOR

(MCPR$_f$)



POWER DEPENDENT MCPR FACTOR
($MCPR_p$)

LINEAR HEAT GENERATION RATE (3.2.3)

The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed:

a. 13.4 kw/ft for the following fuel types:

1. BP8SRB219
2. BP8SRB176

b. 14.4 kw/ft for the following fuel types:

1. BS301E
2. BS301F