October 31, 1986

Dockets Nos. 50-321
and 50-366

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Mr. J. T. Beckham, Jr.
Vice President - Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

The Commission has issued the enclosed Amendments Nos. 132 and 66 to Facility Operating Licenses Nos. DPR-57 and NPF-5, for the Edwin I. Hatch Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 15, as supplemented July 25, and September 22, 1986.

The amendments: (a) permit use of Banked Position Withdrawal Sequences for the first 50 percent of control rod withdrawal; (b) remove the linear mass restriction of 15.2 grams of Uranium-235 per centimeter for fuel assemblies stored in the fuel pool; (c) eliminate specific mechanical descriptions of fuel assemblies; (d) provide Maximum Average Planar Linear Heat Generation limit curves for several new fuel assemblies; and, (e) make several editorial changes.

A copy of our Safety Evaluation and Notice of Issuance are enclosed.

Also enclosed is a copy of an Environmental Assessment and Finding of No Significant Impact which has been published in the Federal Register.

Sincerely,

George W. Rivenbark, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 132 to DPR-57
2. Amendment No. 66 to NPF-5
3. Safety Evaluation
4. Environmental Assessment

cc w/enclosures:
See next page

*Previously Concurred

BWR:PD#2	BWR:PD#2	BWR:PD#2	OGC*
SNorris*	GRivenbark:lk*	DMuller*	
10/6/86	10/6/86	10/22/86	10/08/86

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PDR ADOCK 05000321
P PDR

Mr. J. T. Beckham, Jr.
Georgia Power Company

Edwin I. Hatch Nuclear Plant,
Units Nos. 1 and 2

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 132
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated April 15, 1986, as supplemented July 25 and September 22, 1986 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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;

   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 Facility Operating License No. DPR-57 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 132, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 31, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 132

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

Remove	Insert
x	x
---	xi
3.3-5	3.3-5
3.3-6	3.3-6
3.3-7	3.3-7
3.3-16	3.3-16
3.3-17	3.3-17
3.10-7	3.10-7
3.11-1	3.11-1
3.11-2	3.11-2
3.11-3	3.11-3
3.11-4a	3.11-4a
Figure 3.11-1 (Sheet 1)	Figure 3.11-1 (Sheet 1)
Figure 3.11-1 (Sheet 2)	Figure 3.11-1 (Sheet 2)
Figure 3.11-1 (Sheet 3)	Figure 3.11-1 (Sheet 3)
Figure 3.11-1 (Sheet 4)	Figure 3.11-1 (Sheet 4)
Figure 3.11-1 (Sheet 5)	Figure 3.11-1 (Sheet 5)
Figure 3.11-1 (Sheet 6)	Figure 3.11-1 (Sheet 6)
Figure 3.11-1 (Sheet 7)	Figure 3.11-1 (Sheet 7)
---	Figure 3.11-1 (Sheet 8)
Figure 3.11-2 (Sheet 1)	Figure 3.11-2
Figure 3.11-4	Figure 3.11-4
Figure 3.11-5	Figure 3.11-5
Figure 3.11-6	Figure 3.11-6
Figure 3.11-7	---
5.0-1	5.0-1
5.0-2	5.0-2

LIST OF FIGURES

LIST OF FIGURES (Continued)

3.3.F. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

A Limiting Rod Pattern for RWE exists when:

1. Thermal power is below 90% of rated and the MCPR is less than 1.70, or

2. Thermal power is 90% of rated or above and the MCPR is less than 1.40.

During operation with a Limiting Control Rod Pattern for RWE and when core thermal power is $\geq$ 30%, either:

1. Both RBM channels shall be operable, or

2. If only one RBM channel is operable, control rod withdrawal shall be blocked within 24 hours, or

3. If neither RBM channel is operable, control rod withdrawal shall be blocked.

G. Limiting the Worth of a Control Rod Below 20% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Whenever the reactor is in the Start & Hot Standby or Run Mode below 20% rated thermal power, the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

4.3.F. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

During operation when a Limiting Control Rod Pattern for RWE exists and only one RBM channel is operable, an instrument functional test of the RBM shall be performed prior to withdrawal of the control rod(s). A Limiting Rod Pattern for RWE is defined by 3.3.F.

G. Limiting the Worth of a Control Rod Below 20% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Prior to the start of control rod withdrawal at startup, and as soon as automatic initiation of the RWM occurs during rod insertion while shutting down, the capability of the Rod Worth Minimizer to properly fulfill its function shall be verified by the following checks.

a. The correctness of the Banked Position Withdrawal Sequence input to the RWM computer shall be verified.

b. The RWM computer on line diagnostic test shall be successfully performed.

c. Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

d. The rod block function of the RWM shall be verified by withdrawing or inserting an out-of-sequence control rod no more than to the block point.

3.3.G.2. Rod Sequence Control System (RSCS)

a. Operability

When the reactor is in the Start and Hot Standby or Run Mode below 20% rated thermal power and control rod movement is within the group notch mode after 50% of the control rods have been withdrawn, the Rod Sequence Control System shall be operable except when performing the RWM surveillance tests.

b. Failed Position Switch

Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System if the actual rod position is known. These rods shall be moved in sequence to their correct positions (full in on insertion or full out on withdrawal).

2. Rod Sequence Control System (RSCS)

a. Operability

As soon as the group notch mode is entered during each reactor startup and as soon as automatic initiation of the RSCS occurs during rod insertion while shutting down, the capability of the Rod Sequence Control System to properly fulfill its function shall be verified by attempting to select and move a rod in each of the out-of-sequence groups.
When the control rod movement is within the group notch mode and as soon as automatic initiation of the RSCS occurs during rod insertion while shutting down, the operability of the notching restriction shall be demonstrated by attempting to move a control rod more than one notch in the first programmed rod group.

b. Failed Position Switch

A second licensed operator shall verify the conformance to Specification 3.3.G.2.b before a rod may be bypassed in the Rod Sequence Control System.

3.3.G.2.c. Shutdown Margin/Scram Time Testing

In order to perform the required shutdown margin demonstrations subsequent to any fuel loading operations, or to perform control rod drive scram and/or friction testing as specified in Surveillance Requirement 4.3.C.2 and the initial startup test program, the relaxation of the following RSCS restraints is permitted. The sequence restraints imposed on control rod groups A_{12}, A_{34}, B_{12}, or B_{34} after 50% of the control rods have been withdrawn may be removed for the test period by means of the individual rod position bypass switches.

4.3.G.2.c. Shutdown Margin/Scram Time Testing

Prior to control rod withdrawal for startup, verify the conformance to Specification 3.3.G.2.b. before a rod may be bypassed in the RSCS. The requirements to allow use of the individual rod position bypass switches within rod groups A_{12}, A_{34}, B_{12}, or B_{34} of the RSCS during shutdown margin, scram time or friction testing are:

(1) RWM operable as per Specification 3.3.G.1.

(2) After the bypassing of the rods in the RSCS groups A_{12}, A_{34}, B_{12}, or B_{34} for test purposes, it shall be demonstrated that movement of the rods in the 50% density to the preset power level range is blocked or limited to the single notch mode of withdrawal.

(3) A second licensed operator shall verify the conformance to procedures and this Specification.

H. Shutdown Requirements

If Specifications 3.3.A through 3.3.G are not met, an orderly shutdown shall be initiated and the reactor placed in the Cold Shutdown Condition within 24 hours.

3.3.G.1. Rod Worth Minimizer (RWM) (Continued)

In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

At power levels below 20% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range of RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator or other qualified technical plant employee whose qualifications have been reviewed by the AEC can manually fulfill the control rod pattern conformance functions of this system.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no consequences are acceptable. Control rod pattern constraints above 20% of rated power are imposed by power distribution requirements as defined in Section 3.11 and 4.11 of these Technical Specifications. Power level for automatic cutout of the RSCS function in sensed by first stage turbine pressure. Because the instrument has an instrument error of ± 10% of full power the nominal instrument setting is 30% of rated power. Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow and is set nominally at 30% of rated power to be consistent with the RSCS setting.

Surveillance Requirements:

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 20% of rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

2. Rod Sequence Control System (RSCS)

a. Operability

Limiting Conditions for Operation:

See bases for Technical Specification 3.3.G.1. Rod Worth Minimizer.

3.3.G.2.a. Operability

Surveillance Requirements:

The RSCS can be functionally tested after 50% of the control rods have been withdrawn, by demonstrating that the continuous withdrawal mode for the control drives is inhibited.

This demonstration is made by attempting to withdraw a control rod more than one notch in the first programmed rod group subsequent to reaching the 50% rod density point. This restriction to the notching mode of operation for control rod withdrawal is automatically removed when the reactor reaches the automatic initiation setpoint.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

b. Failed Position Switch

Limiting Conditions for Operation:

In the event that a control rod has a failed "Full-in" or "Full-out" position switch, it may be bypassed in the Rod Sequence Control System if its position is otherwise known. It is a safer and more desirable condition for such rods to occupy their proper positions in the control rod patterns during reactor startup or shutdown.

Surveillance Requirements:

Having a second licensed operator verify the actual rod position prior to bypassing a rod in the Rod Sequence Control System provides assurance that Specification 3.3.G.2.b. is met.

c. Shutdown Margin/Scram Time Testing

After initial fuel loading and subsequent refuelings when operating above 950 psig all control rods shall be scram tested within the constraints imposed by the RSCS and before the 40% power level is reached. To maintain the required reactor pressure conditions the individually scrammed or inserted rod should be withdrawn to its original position immediately following testing of each rod. In order to select and withdraw the scrammed or inserted insequence control rod (also to select and insert a fully withdrawn insequence rod in case of friction testing) it will be necessary to simulate all the insequence withdrawn rods of the succeeding RSCS groups as being at full in position by utilizing the individual rod posi-

3.10.A.2. Fuel Grapple Hoist Load Setting Interlocks

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 1500 lbs. in comparison to the load setting of 485 ± 30 lbs.

3. Auxiliary Hoists Load Setting Interlock

Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 485 ± 30 lb load setting of these hoists is adequate to trip the interlock when a fuel bundle is being handled.

B. Fuel Loading

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

C. Core Monitoring During Core Alterations

The SRM's are provided to monitor the core during periods of Unit shutdown and to guide the operator during refueling operations and Unit startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirements of 3 counts per second provides assurance that neutron flux is being monitored.

During spiral unloading, it is not necessary to maintain 3 cps because core alterations will involve only reactivity removal and will not result in criticality.

The loading of up to four fuel bundles around the SRM's before attaining the 3 cps is permissible because these bundles were in a subcritical configuration when they were removed and therefore they will remain subcritical when placed back in their previous positions.

D. Spent Fuel Pool Water Level

The design of the spent fuel storage pool provides a storage location for 3181 fuel assemblies in the reactor building which ensures adequate shielding, cooling, and the reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the active fuel will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the water level to less than 10 feet above the top of the active fuel. Lines extending below this level are equipped with two check valves in series to prevent inadvertent pool drainage. All fuel loaded into the Edwin I. Hatch Nuclear Plant spent fuel pool shall have an uncontrolled lattice k_∞ less than or equal to the limit for high density fuel racks described in the "General Electric Standard Application for Reactor Fuel" (GESTAR II), NEDE-24011-P-A-8. Alternatively, fuel not described in GESTAR II shall have been analyzed with another NRC approved methodology to ensure conformity to the FSAR design basis for fuel in the spent fuel racks.

E. Control Rod Drive Maintenance

During certain periods, it is desirable to perform maintenance on two control rod drives at the same time.

3.11. FUEL RODS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for all core locations shall not exceed the appropriate APLHGR limit for those core locations. The APLHGR limit, which is a function of average planar exposure and fuel type, is the appropriate value from Figure 3.11-1, sheets 1 through 6, multiplied by the smaller of the two MAPFAC factors determined from Figure 3.11-1, sheets 7 and 8. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

B. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR as a function of core height shall not exceed the limiting value shown in Figure 3.11-2 for 7 x 7 fuel or the limiting value of 13.4 kw/ft for any 8 x 8 fuel. If at any time during

4.11. FUEL RODS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥ 25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

The LHGR as function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.

3.11.B. Linear Heat Generation Rate (LHGR) (Continued)

operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

C. Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR), which is a function of scram time, core power, and core flow. For 25% ≤ power < 30%, the OLMCPR is given in Figure 3.11.6. For power ≥ 30%, the OLMCPR is the greater of either:

1. The applicable limit determined from Figure 3.11.3, or

2. The applicable limit from either Figures 3.11.4 or 3.11.5, multiplied by the K_p factor determined from Figure 3.11.6, where:

$\tau = 0$ or $\left[\frac{\tau_{ave}-\tau_B}{\tau_A-\tau_B}\right]$, whichever is greater

τ_A = 0.90 sec (Specifications 3.3.C.2.a. scram time limit to 20% insertion from fully withdrawn)

$$\tau_B = 0.710+1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2}(0.053)\ \text{[Ref.10]}$$

4.11.C.1. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined to be equal to or greater than the applicable limit, daily during reactor power operation at ≥ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.F.

4.11.C.2. Minimum Critical Power Ratio Limit

The MCPR limit at rated flow and rated power shall be determined for each fuel type, 8X8R, P8X8R, BP8x8R or 7X7 from figures 3.11.4 and 3.11.5 respectively using:

a. τ=1.0 prior to initial scram time measurements for the cycle, performed in accordance with specifications 4.3.C.2.a.

or

b. τ as defined in specification 3.11.C.

The determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by specification 4.3.C.2.

3.11. FUEL RODS

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K, even considering the postulated effects of fuel pellet densification.

The peak cladding temperature 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 only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures conform to 10 CFR 50.46. The limiting value for APLHGR at rated conditions is shown in Figures 3.11.1. sheets 1 thru 6.

A flow dependent correction factor incorporated in to Figure 3.11-1 (sheet 8) is applied to the rated conditions APLHGR to assure that the 2200°F PCT limit is complied with during LOCA initiated from less than rated core flow. In addition, other power and flow dependent corrections given in Figure 3.11-1 (sheets 7 and 8) are applied to the rated conditions APLHGR limits to assure that the fuel thermal-mechanical design criteria are met during abnormal transients initiated from off-rated conditions.

The calculational procedure used to establish the APLHGR shown in Figures 3.11.1, sheets 1 thru 6, is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis as compared to previous analyses performed with Reference 1 are: (1) The analyses assume a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figure 3.11.1; (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting as described in Reference 2, are included in the reflooding calculations.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 1 of NEDO-21187[3]. Further discussion of the APLHGR bases is found in NEDC-30474-p[11].

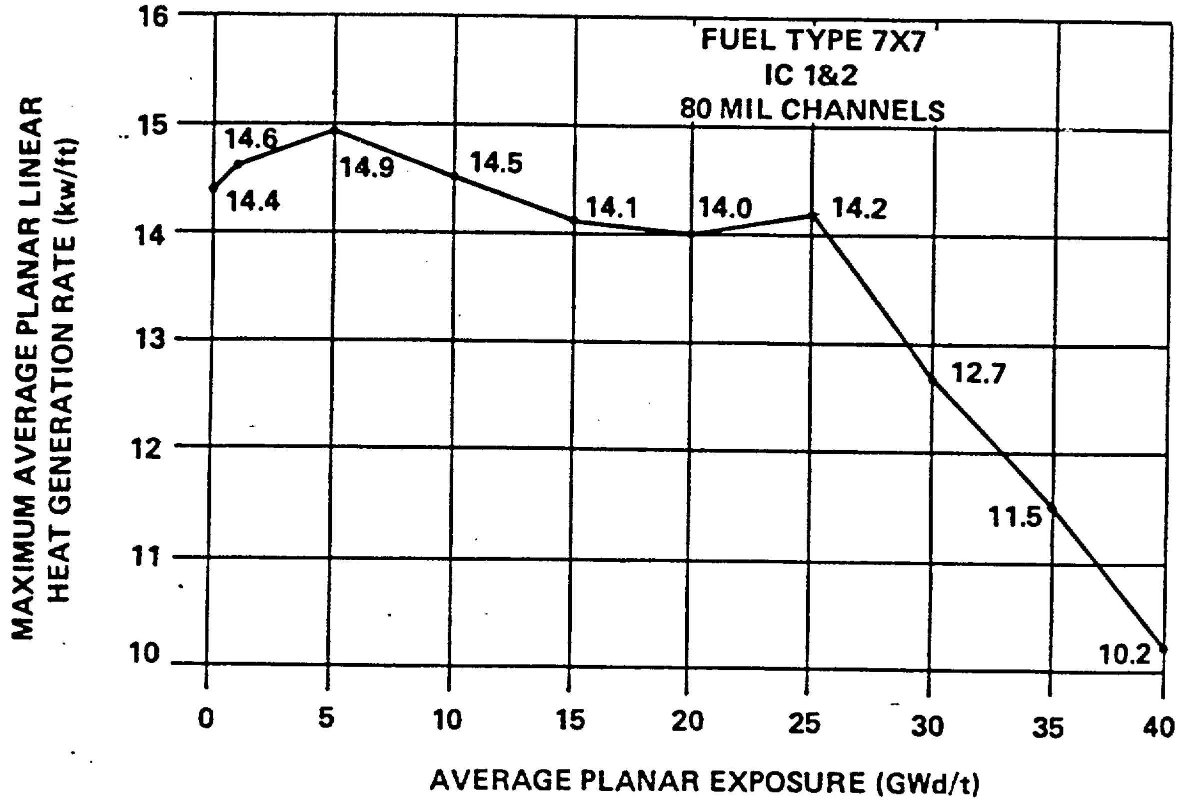
3.11.C. Minimum Critical Power Ratio (MCPR) (Continued)

The purpose of the $MCPR_f$, and the K_p of Figures 3.11.3 and 3.11.6, respectively, 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 $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam 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 the 105% of rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

The core power dependent MCPR operating limit $MCPR_p$ is the power rated flow MCPR operating limit multiplied by the K_p factor given in Figure 3.11.6.

The K_ps are established to protect the core from transients other than core flow increases, including the localized event such as rod withdrawal error. The K_ps were determined based upon the most limiting transient at the given core power level. (For further information on MCPR operating limits for off-rated conditions, reference NEDC-30474-P.[11])



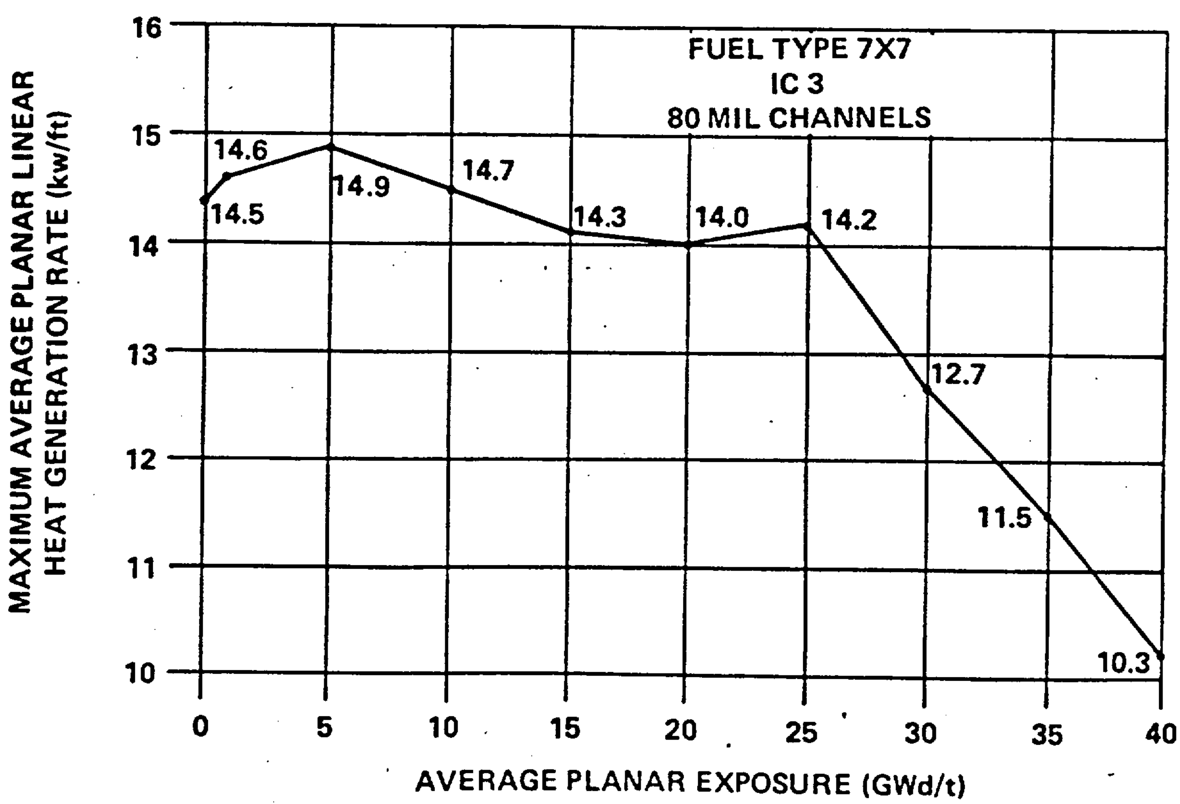
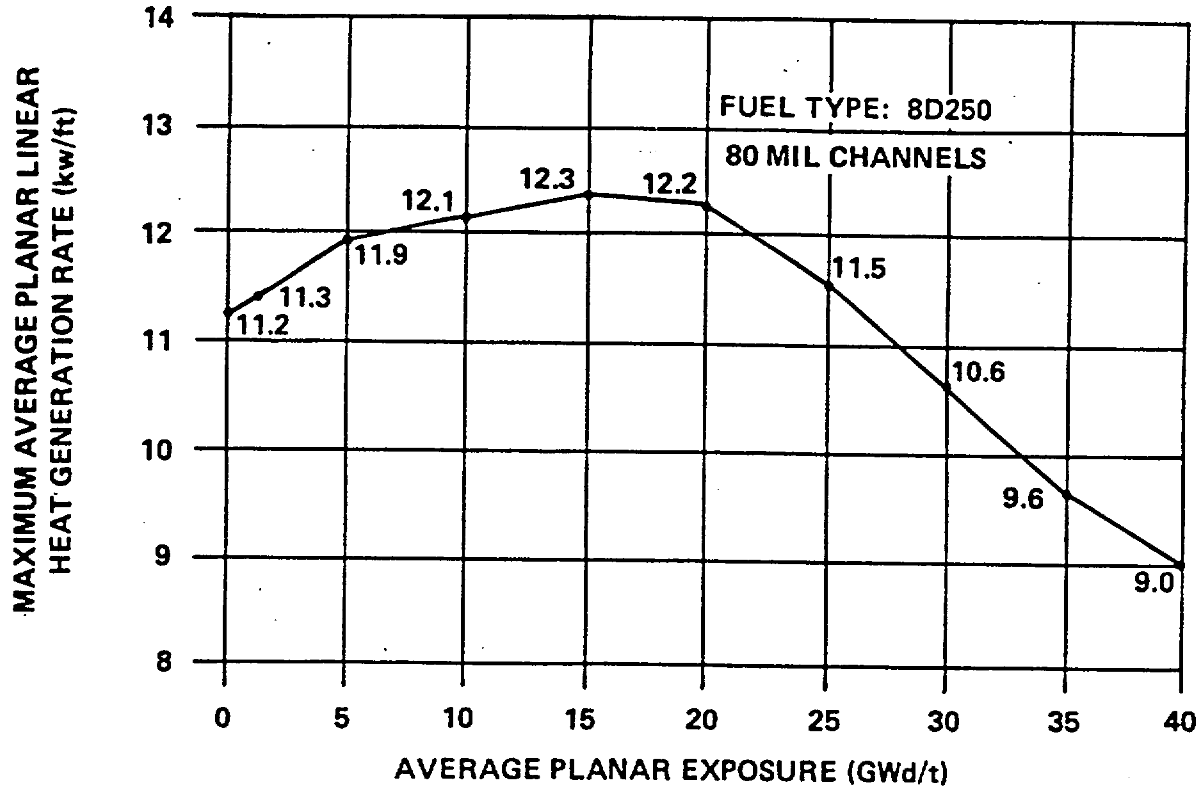


FIGURE 3.11-1 (SHEET 1)



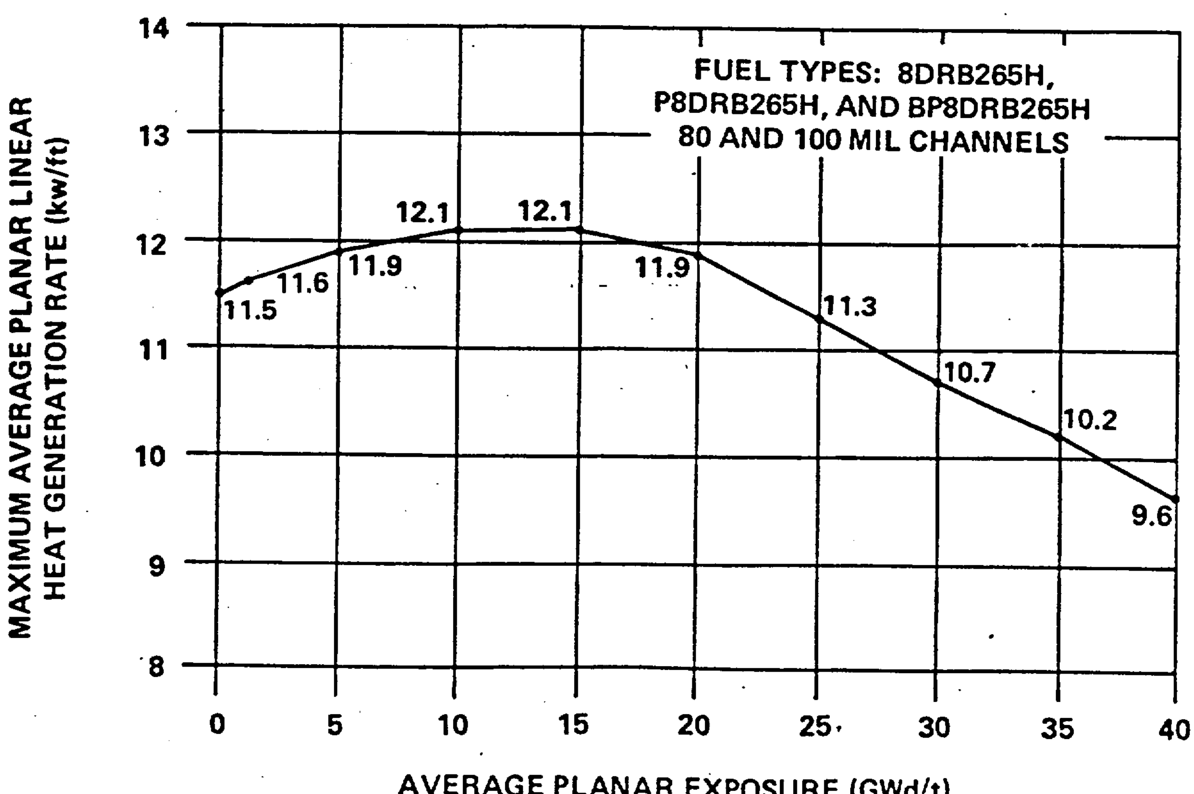
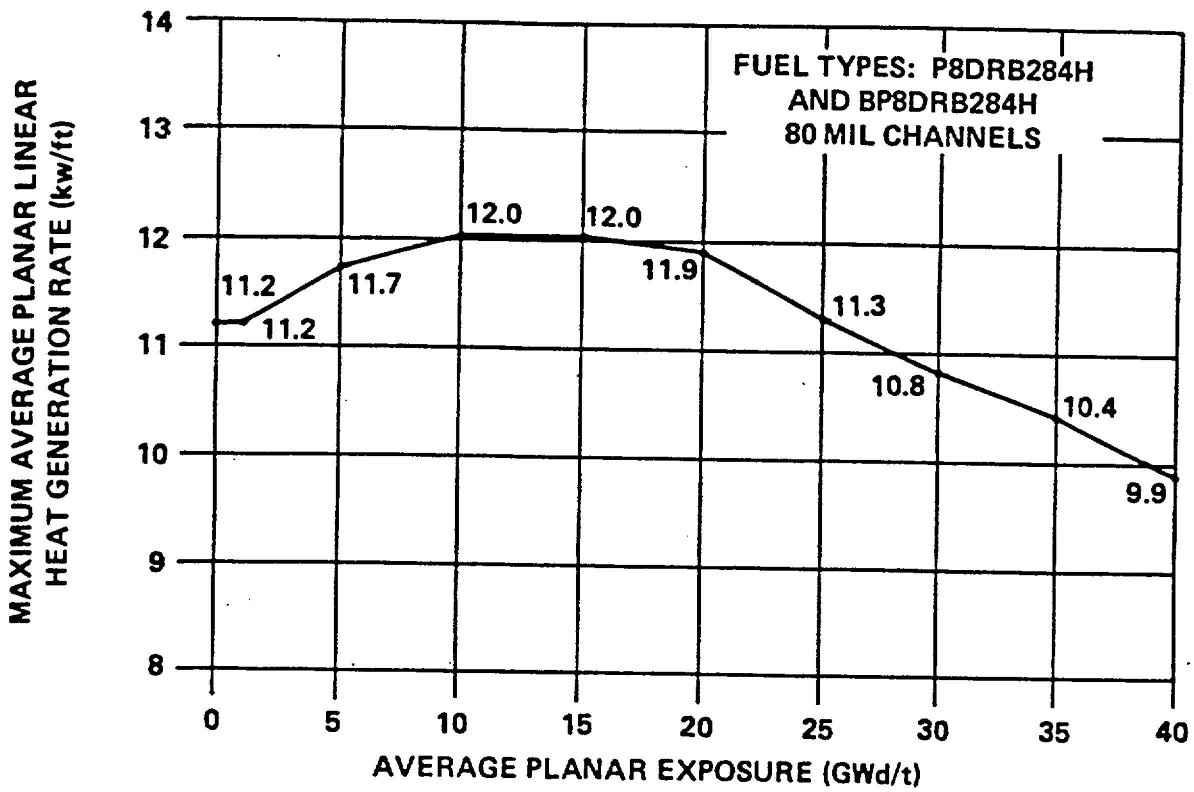


FIGURE 3.11-1 (SHEET 2)



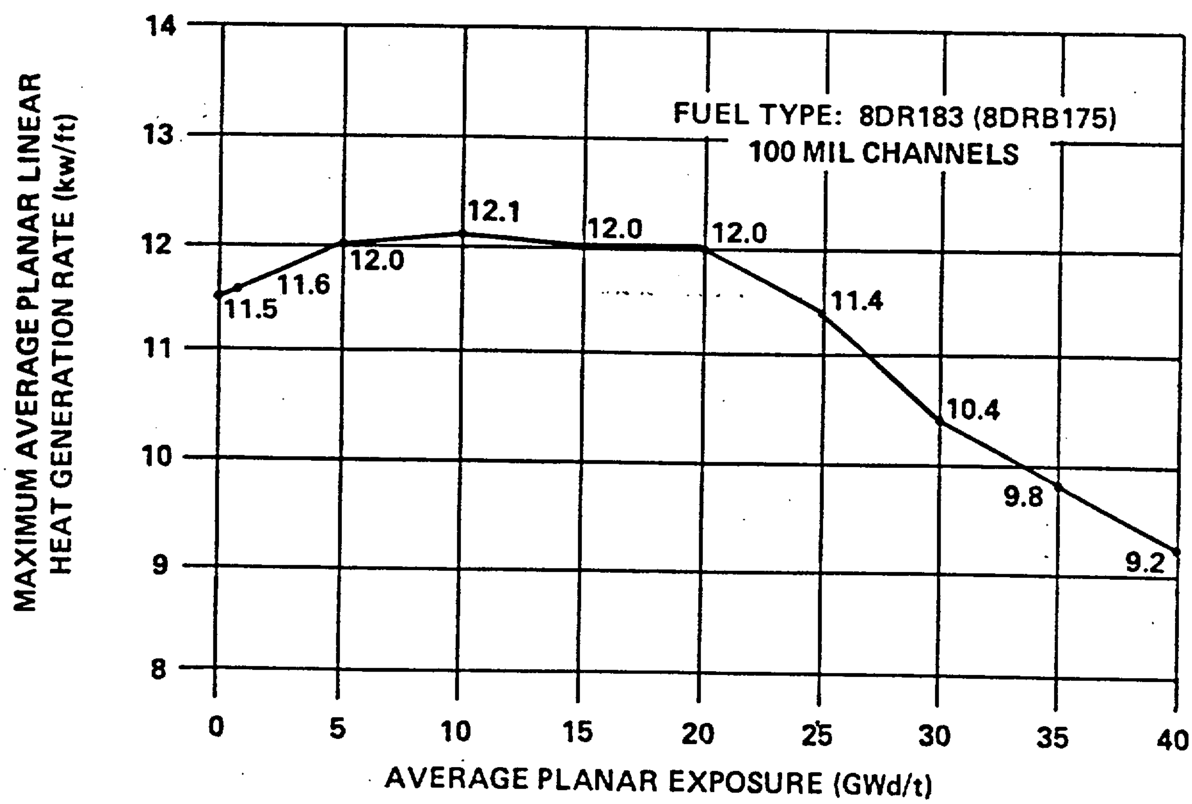
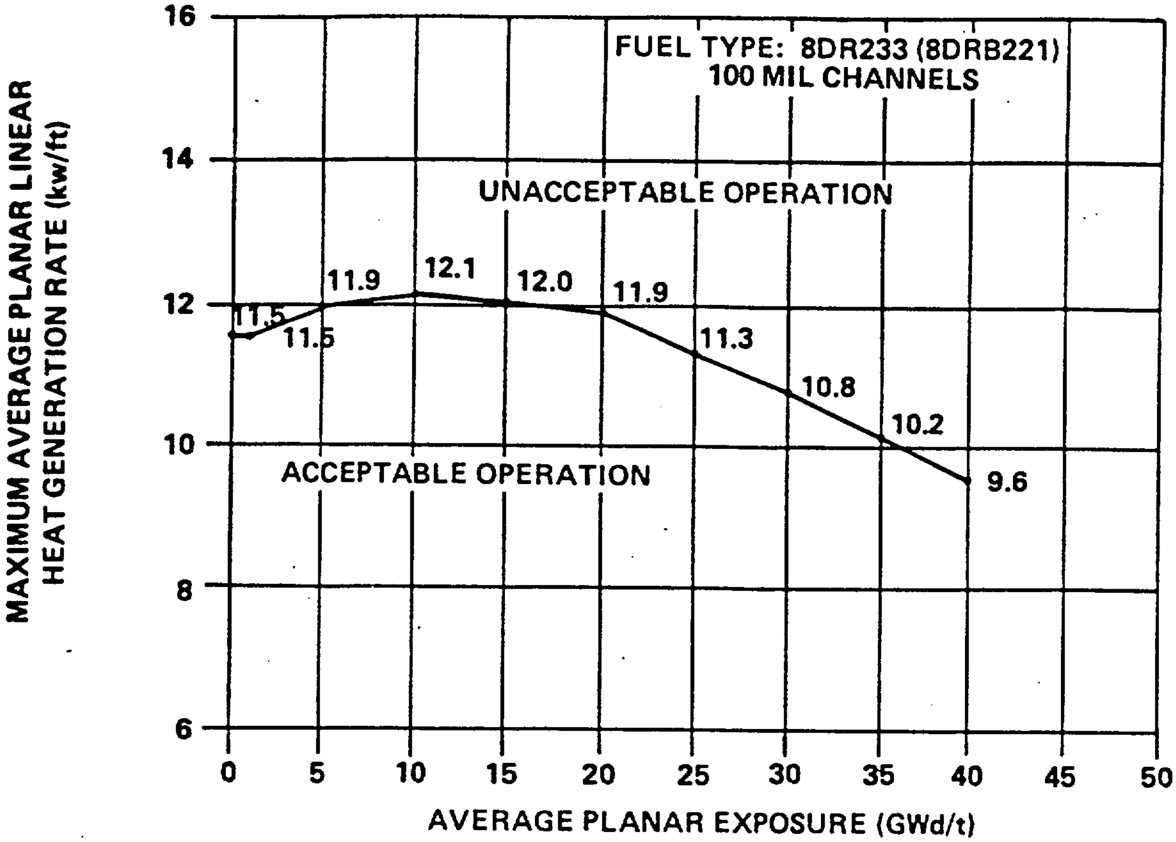


FIGURE 3.11-1 (SHEET 3)



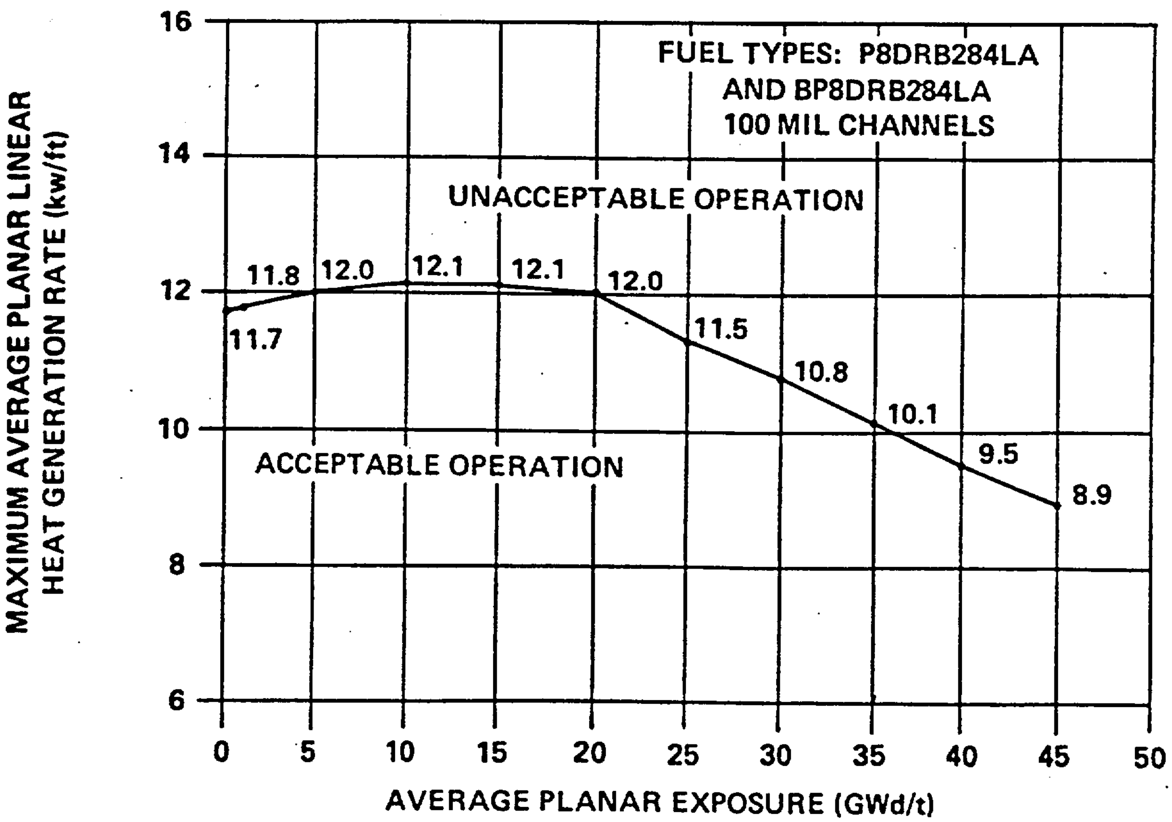
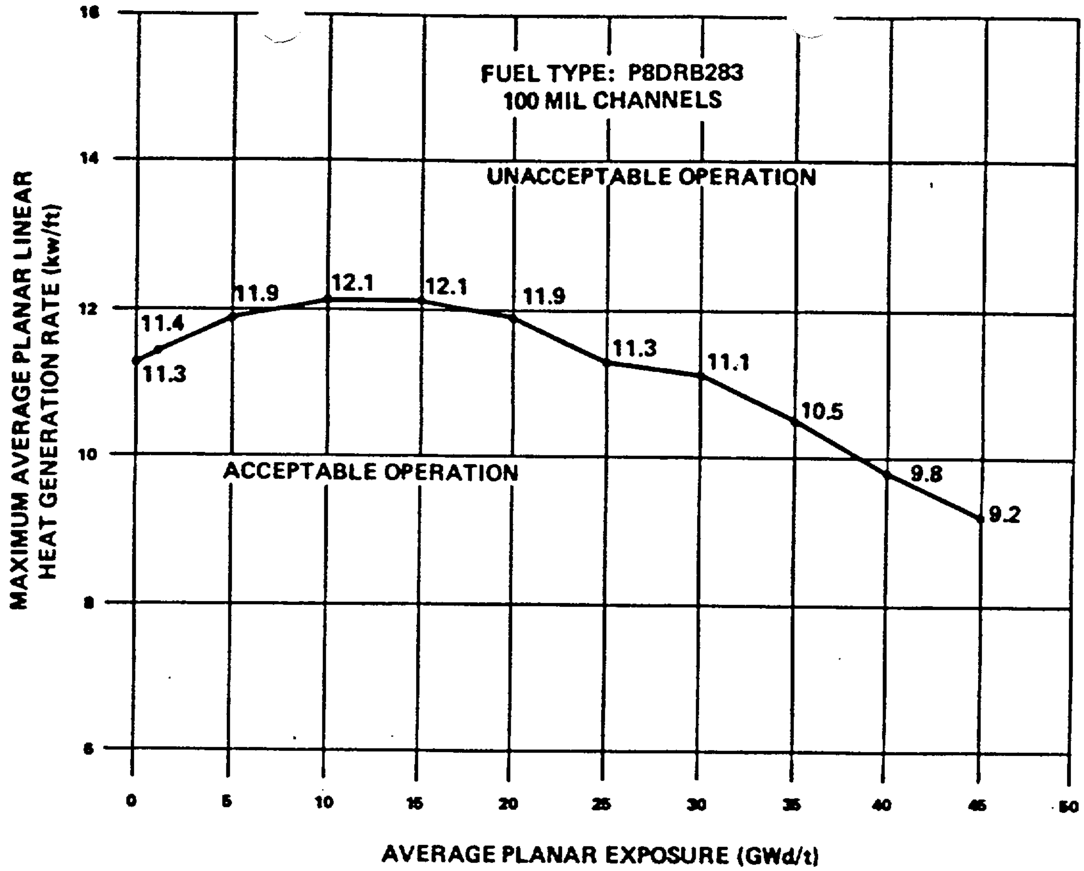


FIGURE 3.11-1 (SHEET 4)



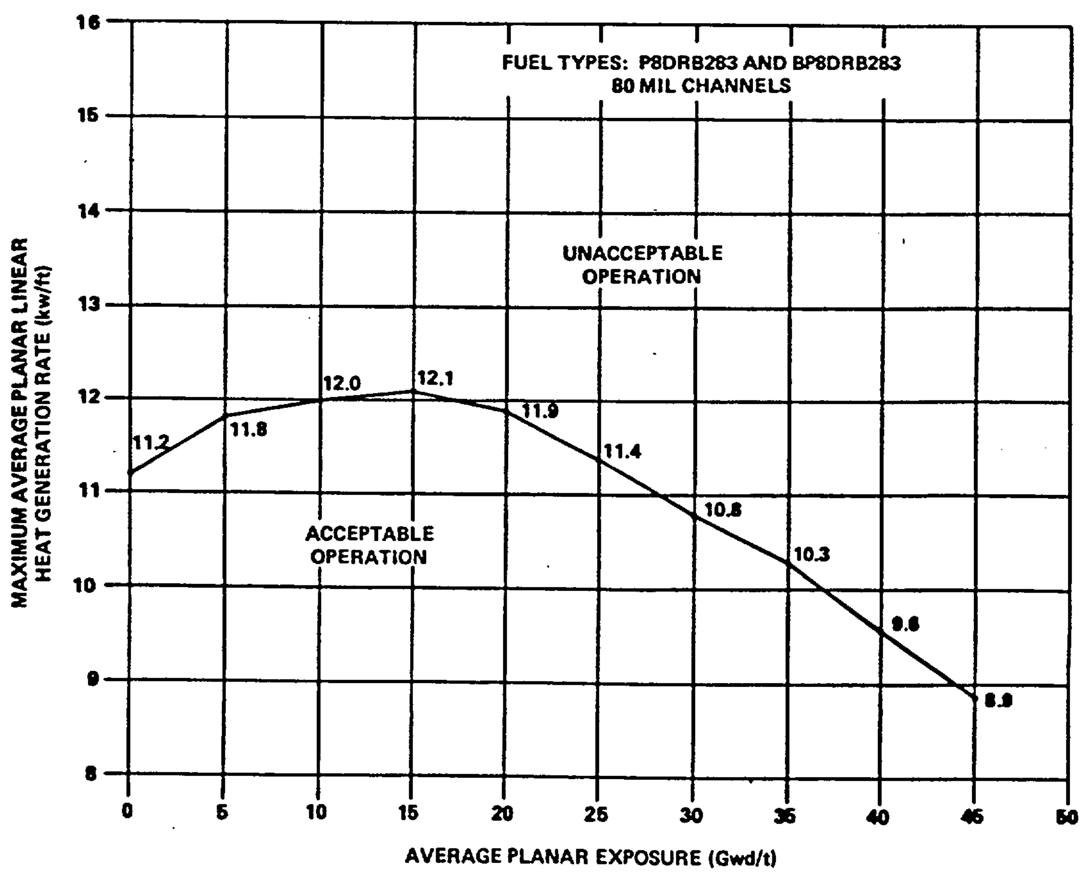


FIGURE 3.11-1 (SHEET 5)

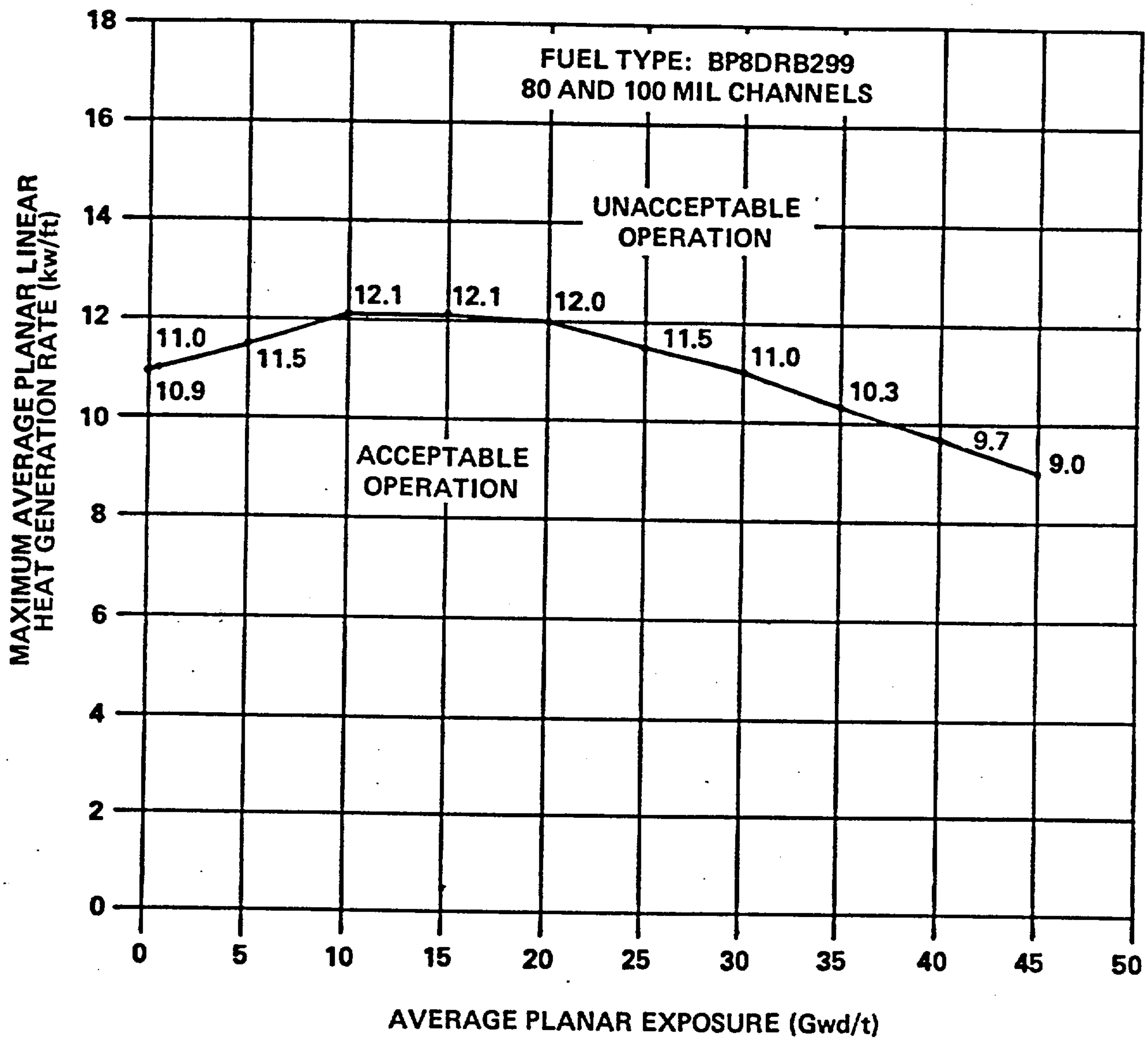


FIGURE 3.11-1 (SHEET 6)

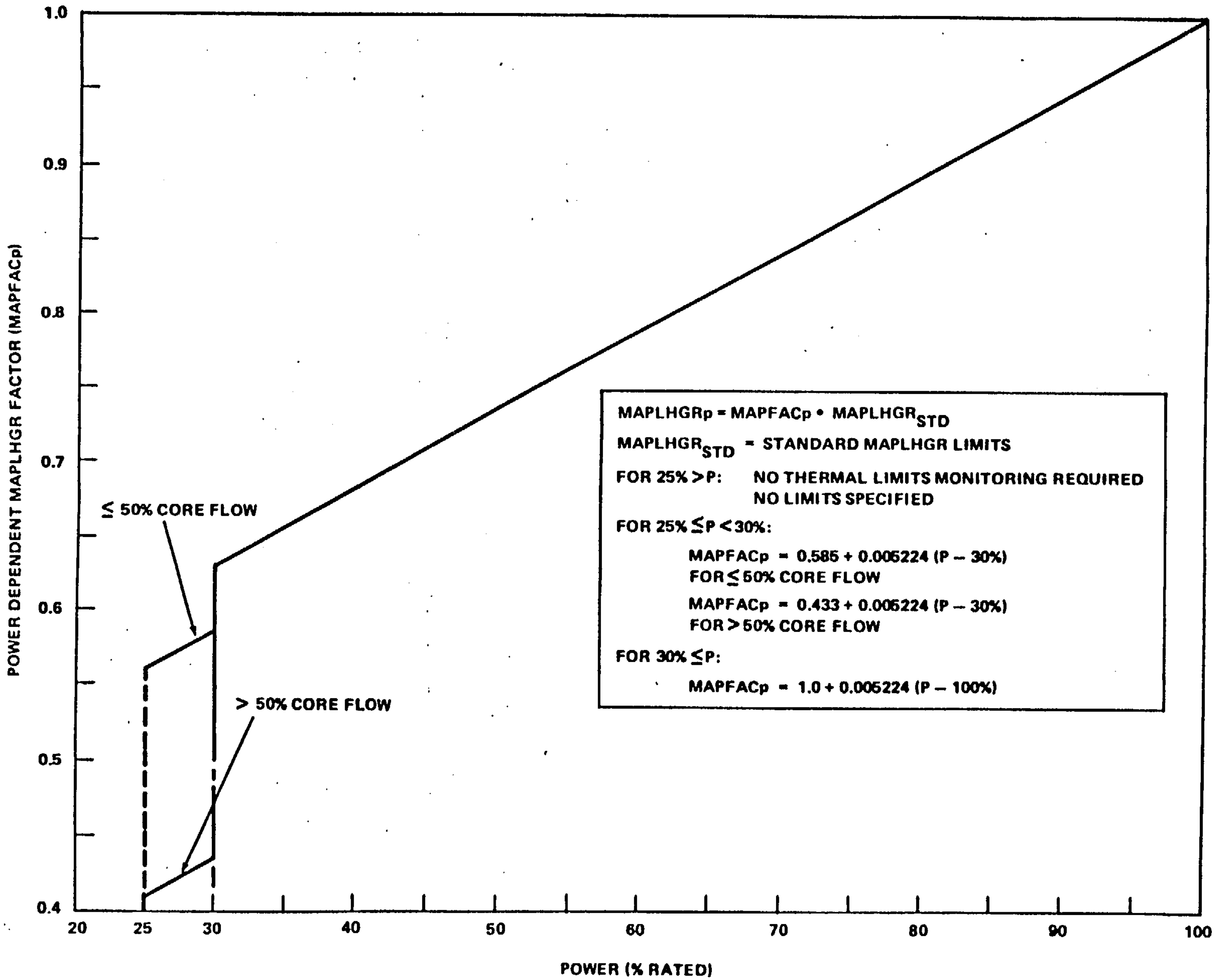
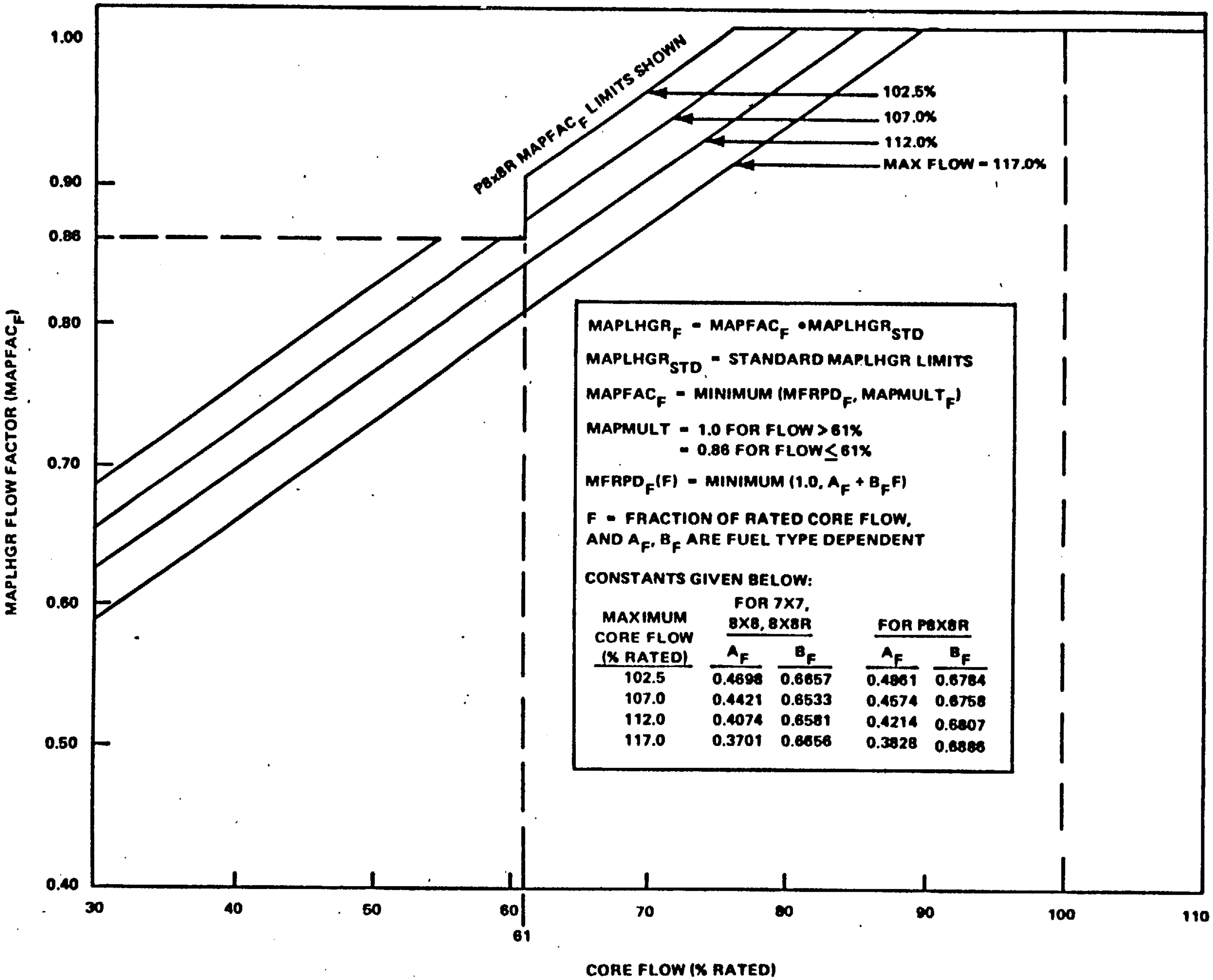


FIGURE 3.11-1 (SHEET 7) MAPFACp



$MAPLHGR_F$ = $MAPFAC_F$ • $MAPLHGR_{STD}$

$MAPLHGR_{STD}$ = STANDARD MAPLHGR LIMITS

$MAPFAC_F$ = MINIMUM ($MFRPD_F$, $MAPMULT_F$)

MAPMULT = 1.0 FOR FLOW > 61%
= 0.86 FOR FLOW ≤ 61%

$MFRPD_F(F)$ = MINIMUM (1.0, $A_F + B_F F$)

F = FRACTION OF RATED CORE FLOW,
AND A_F, B_F ARE FUEL TYPE DEPENDENT

CONSTANTS GIVEN BELOW:

MAXIMUM CORE FLOW (% RATED)	FOR 7X7, 8X8, 8X8R		FOR P8X8R	
	A_F	B_F	A_F	B_F
102.5	0.4698	0.6657	0.4861	0.6784
107.0	0.4421	0.6533	0.4574	0.6758
112.0	0.4074	0.6581	0.4214	0.6807
117.0	0.3701	0.6656	0.3828	0.6886

FIGURE 3.11-1 (SHEET 8) $MAPFAC_F$

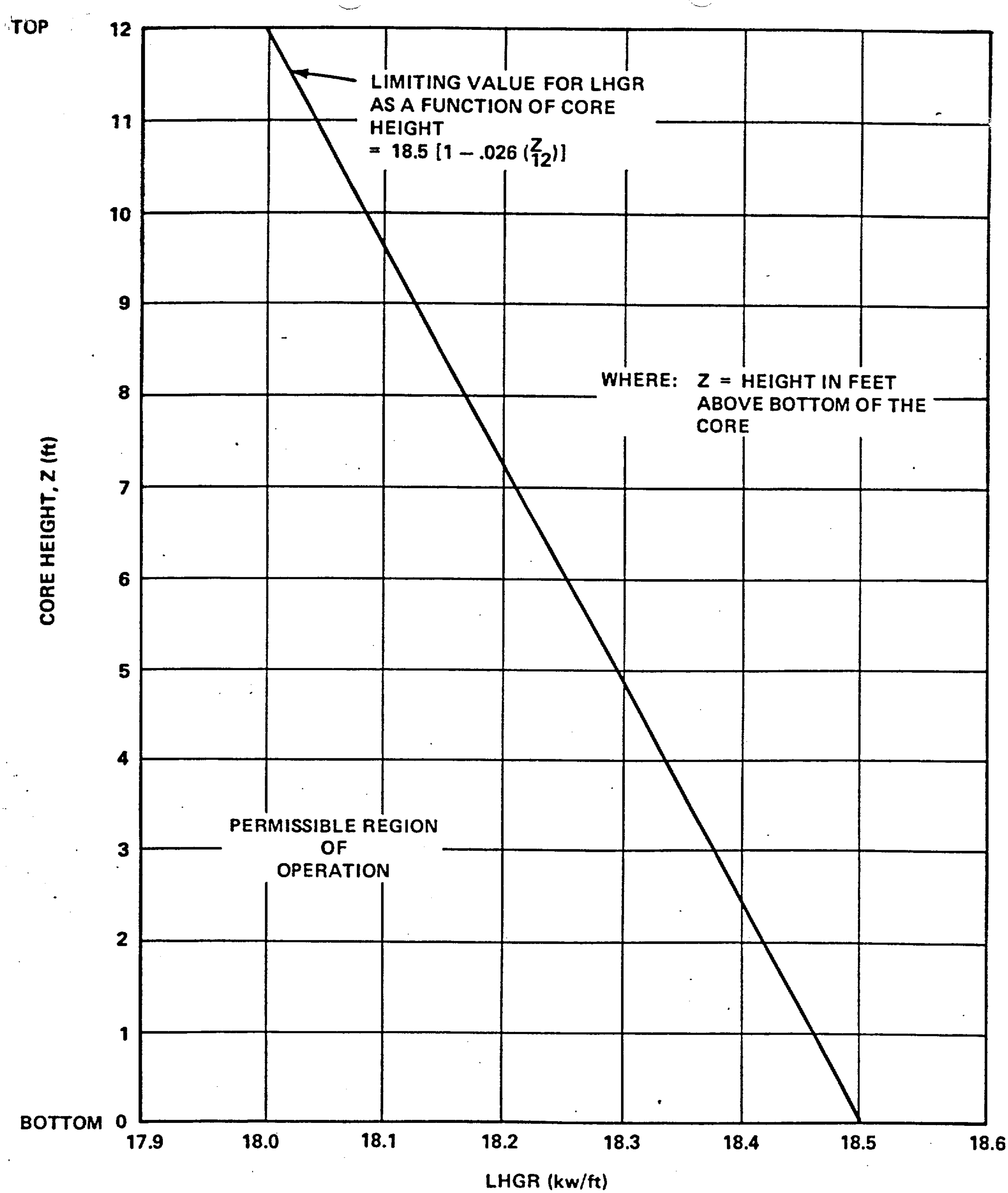


FIGURE 3.11-2
LIMITING VALUE FOR LHGR
FUEL TYPE 7X7

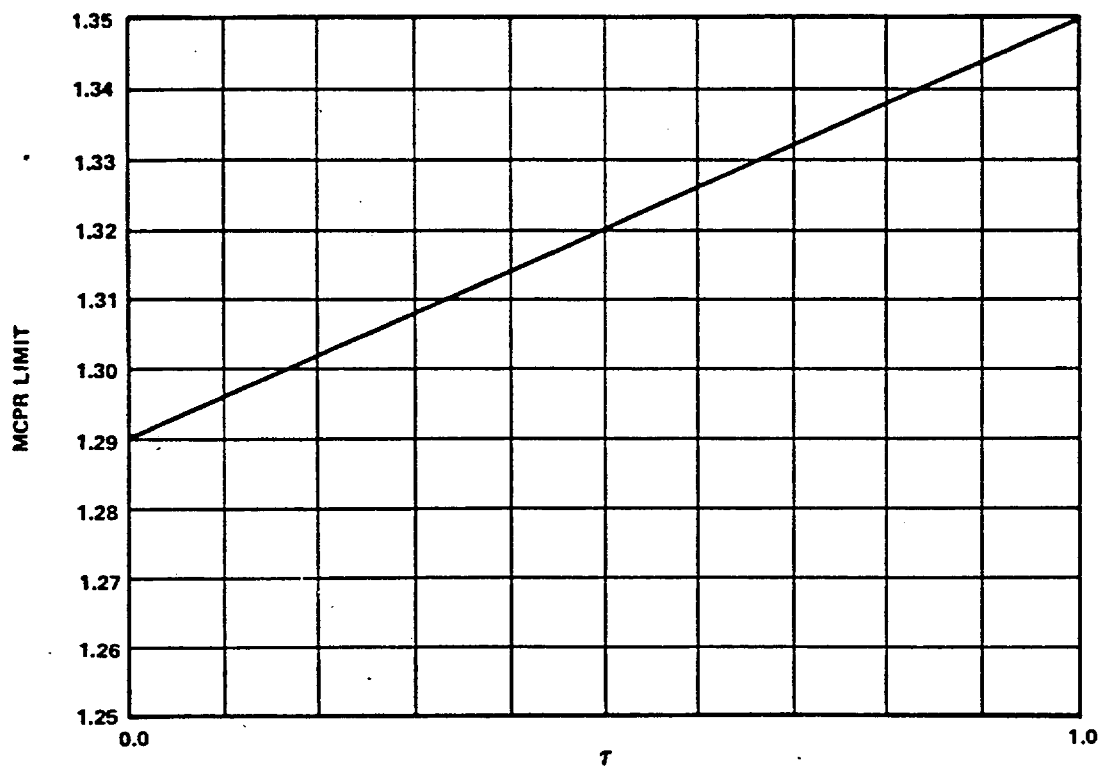


FIGURE 3.11.4
MCPR LIMIT FOR ALL 8X8 FUEL TYPES
FOR RATED POWER AND RATED FLOW

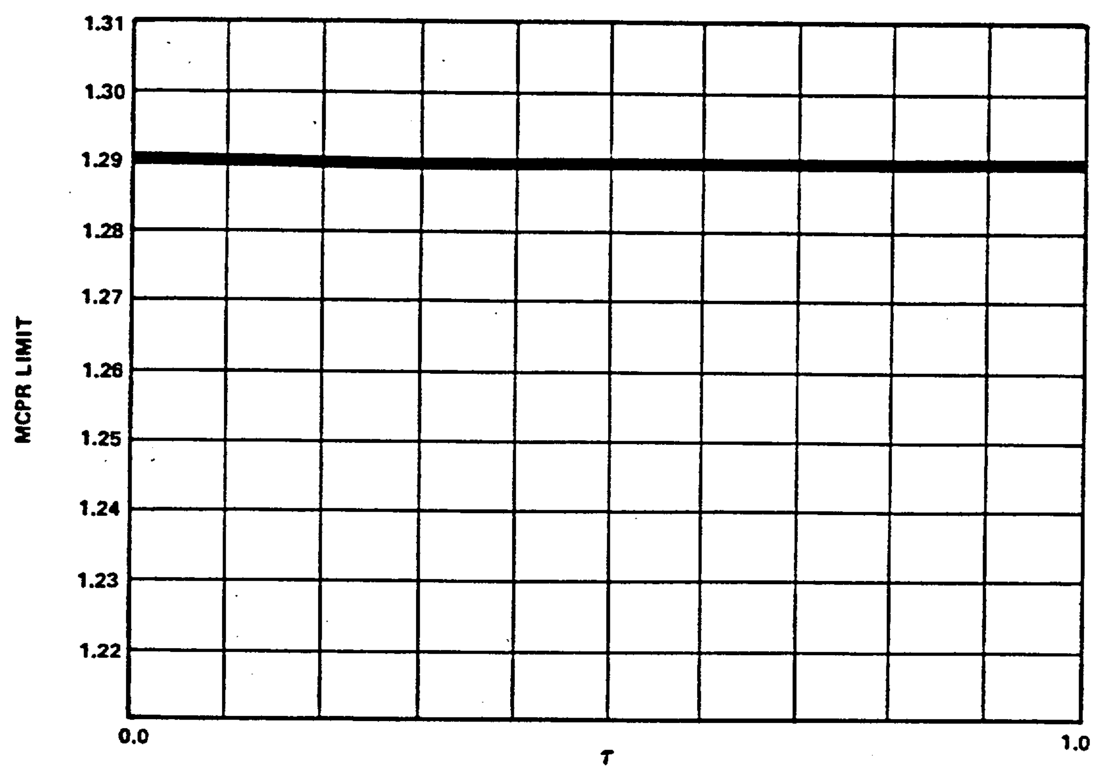


FIGURE 3.11-5
MCPR LIMIT FOR 7X7 FUEL
FOR RATED POWER AND RATED FLOW

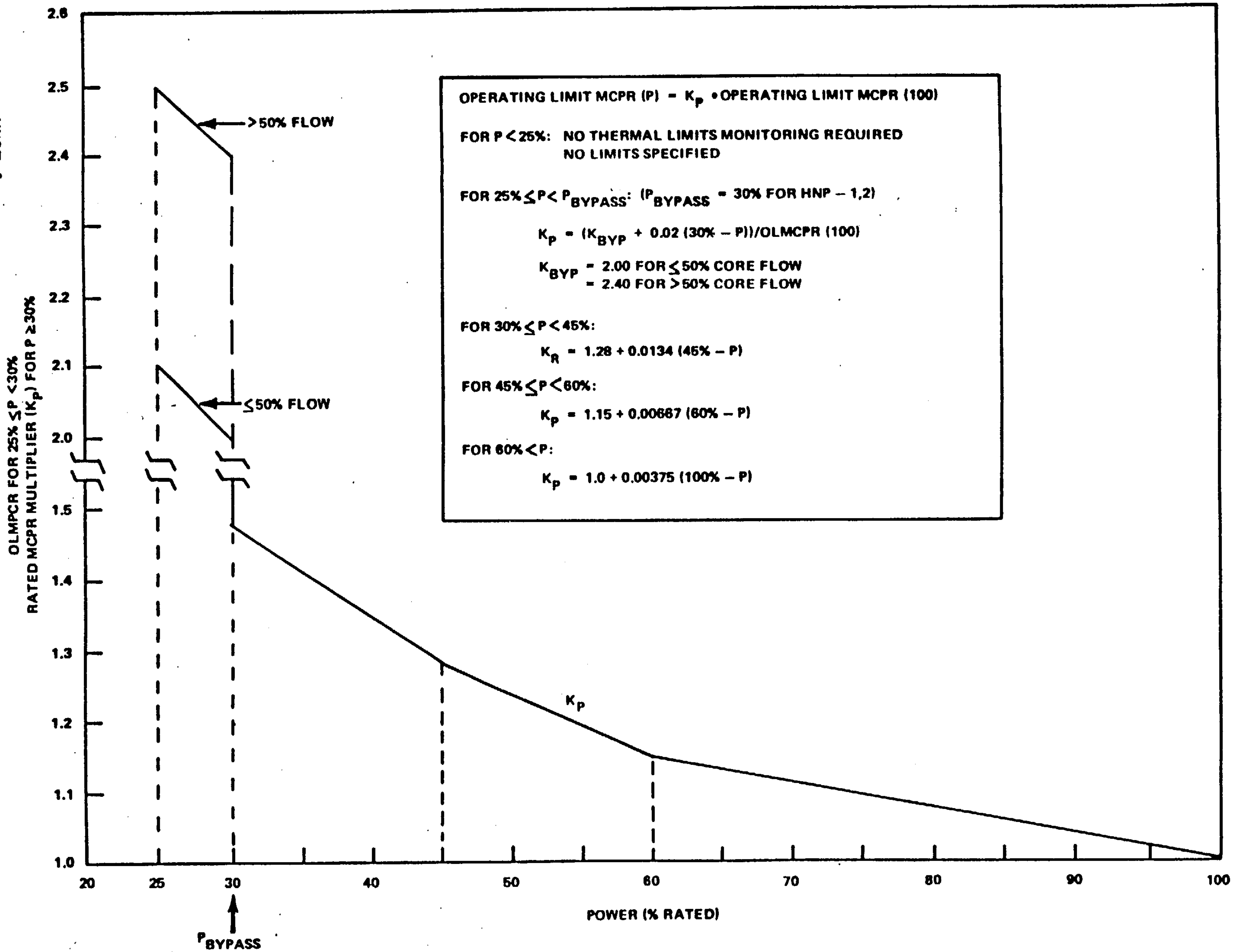


FIGURE 3.11-6 Kp

5.0 MAJOR DESIGN FEATURES

A. Site

Edwin I. Hatch Nuclear Plant Unit No. 1 is located on a site of about 2244 acres, which is owned by Georgia Power Company, on the south side of the Altamaha River in Appling County near Baxley, Georgia. The Universal Transverse Mercator Coordinates of the center of the reactor building are: Zone 17R LF 372,935.2m E and 3,533,765.2m N.

B. Reactor Core

1. Fuel Assemblies

The core shall consist of not more than 560 fuel assemblies 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 Final Safety Analysis Report (FSAR).

2. Control Rods

The reactor shall contain 137 cruciform-shaped control rods.

C. Reactor Vessel

The reactor vessel is described in Table 4.2-2 of the FSAR. The applicable design specifications shall be as listed in Table 4.2-1 of the FSAR.

D. Containment

1. Primary Containment

The principal design parameters are characteristics of the primary containment shall be as given in Table 5.2-1 of the FSAR.

2. Secondary Containment* (See Page 5.0-1a)

The secondary containment shall be as described in Section 5.3.3.1 of the FSAR and the applicable codes shall be as given in Section 12.4.4 of the FSAR.

3. Primary Containment Penetrations

Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.3.4 of the FSAR.

E. Fuel Storage

1. Spent Fuel

All arrangement of fuel in the spent fuel storage racks shall be maintained in a subcritical configuration having a k_{eff} not greater than 0.95.

2. New Fuel

The new fuel storage vault shall be such that the k_{eff} dry shall not be greater than 0.90 and the k_{eff} flooded shall not be greater than 0.95.

5.0.F. Seismic Design

The reactor building and all engineered safeguard systems are designed for the design basis earthquake with a horizontal ground acceleration of 0.15 g. The operating basis earthquake has a horizontal ground acceleration of 0.08 g.

G. Component Cyclic or Transient Limit

The Reactor Pressure Vessel is designed for and shall be maintained within the cyclic or transient limits of Table 5.0.G-1.

H. References

1. FSAR Section 4.2, Reactor Vessel and Appurtenances Mechanical Design
2. FSAR Section 5.2, Primary Containment System
3. FSAR Section 5.3, Secondary Containment System
4. FSAR Section 12.4.4, Governing Codes and Regulations
5. FSAR Section 10.3, Spent Fuel Storage
6. FSAR Section 10.2, New Fuel Storage

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
License No. NPF-5

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated April 15, 1986, as supplemented July 25 and September 22, 1986 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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;

   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 Facility Operating License No. NPF-5 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 66 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 31, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 66

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The overleaf pages are provided for convenience.

Remove	Insert
3/4 1-14	3/4 1-14
3/4 1-15	3/4 1-15
3/4 1-16	3/4 1-16
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 2-4a	3/4 2-4a
3/4 2-4b	3/4 2-4b
3/4 2-4c	3/4 2-4c
3/4 2-4d	3/4 2-4d
3/4 2-4e	3/4 2-4e
3/4 2-4f	3/4 2-4f
3/4 2-4g	3/4 2-4g
3/4 2-4h	3/4 2-4h
3/4 2-4i	3/4 2-4i
---	3/4 2-4j
---	3/4 2-4k
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-7a	3/4 2-7a
3/4 2-7b	3/4 2-7b
3/4 2-7c	3/4 2-7c
3/4 2-7d	3/4 2-7d
3/4 2-8	3/4 2-8
B 3/4 1-3	B 3/4 1-3
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-4	B 3/4 2-4
B 3/4 9-2	B 3/4 9-2
5-1	5-1
5-4	5-4

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 The control rod drive housing support shall be in place.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.8 The control rod drive housing support shall be inspected after reassembly and verified to be in place, prior to startup, any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

REACTIVITY CONTROL SYSTEMS

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The Rod Worth Minimizer (RWM) shall be OPERABLE.

APPLICABILITY: CONDITIONS 1 and 2*, when THERMAL POWER is less than 20% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, the provisions of Specification 3.0.4 are not applicable, operation may continue and control rod movement is permitted provided that a second licensed operator or other qualified member of the technical staff is present at the reactor control console and verifies compliance with the prescribed control rod pattern.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, and in CONDITION 1 when the RWM is initiated during control rod insertion when reducing THERMAL POWER, by:

   1. Verifying proper annunciation of the selection error of at least one out-of-sequence control rod, and

   2. Verifying the rod block function of the RWM by moving an out-of-sequence control rod.

b. By verifying that the Banked Position Withdrawal Sequence input to the RWM computer is correct following any loading of the sequence program into the computer.

*Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.4.2 The Rod Sequence Control System (RSCS) shall be OPERABLE.

APPLICABILITY: CONDITIONS 1* and 2*#, when THERMAL POWER is less than 20% of RATED THERMAL POWER and control rod movement is within the group notch mode after 50% of the control rods have been withdrawn.

ACTION:

With the RSCS inoperable control rod movement shall not be permitted, except by a scram.

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RSCS shall be demonstrated OPERABLE by:

a. Selecting and attempting to move an inhibited control rod:

1. As soon as the group notch mode is entered during each reactor startup, and

2. As soon as the rod inhibit mode is automatically initiated during control rod insertion.

*See Special Test Exception 3.10.2.

#Entry into CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RSCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

SURVEILLANCE REQUIREMENTS (Continued)

b. Attempting to move a control rod more than one notch as soon as the group notch mode is automatically initiated during control rod:

   1. Withdrawal each reactor startup, and

   2. Insertion.

c. Performance of the comparator check of the group notch circuits prior to control rod;

   1. Movement within the group notch mode during each reactor startup, and

   2. Insertion to reduce THERMAL POWER to less than 20% of RATED THERMAL POWER.

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 be equal to or less than the applicable APLHGR limit, which is a function of fuel type and AVERAGE PLANAR EXPOSURE. The APLHGR limit is given by the applicable rated-power, rated-flow limit taken from Figures 3.2.1-1 through 3.2.1-11, multiplied by the smaller of either:

a. The factor given by Figure 3.2.1-12, or

b. The factor given by Figure 3.2.1-13.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figures 3.2.1-1 through 3.2.1-11, as adjusted per Figures 3.2.1-12 and 3.2.1-13, initiate corrective action within 15 minutes and continue corrective action so that the APLHGR meets 3.2.1 within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

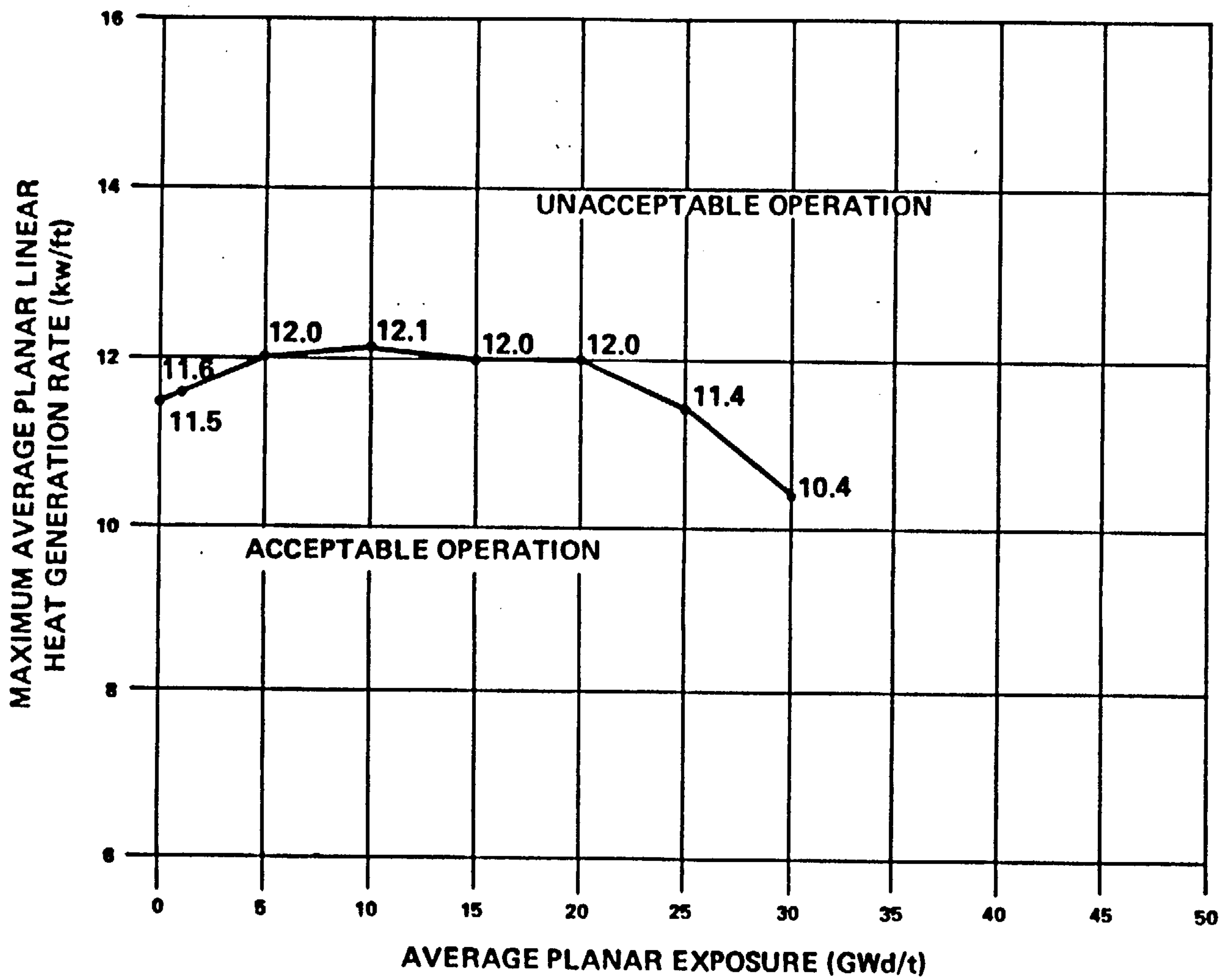
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the applicable limit determined from Figures 3.2.1-1 through 3.2.1-11, as adjusted per Figure 3.2.1-12 and 3.2.1-13:

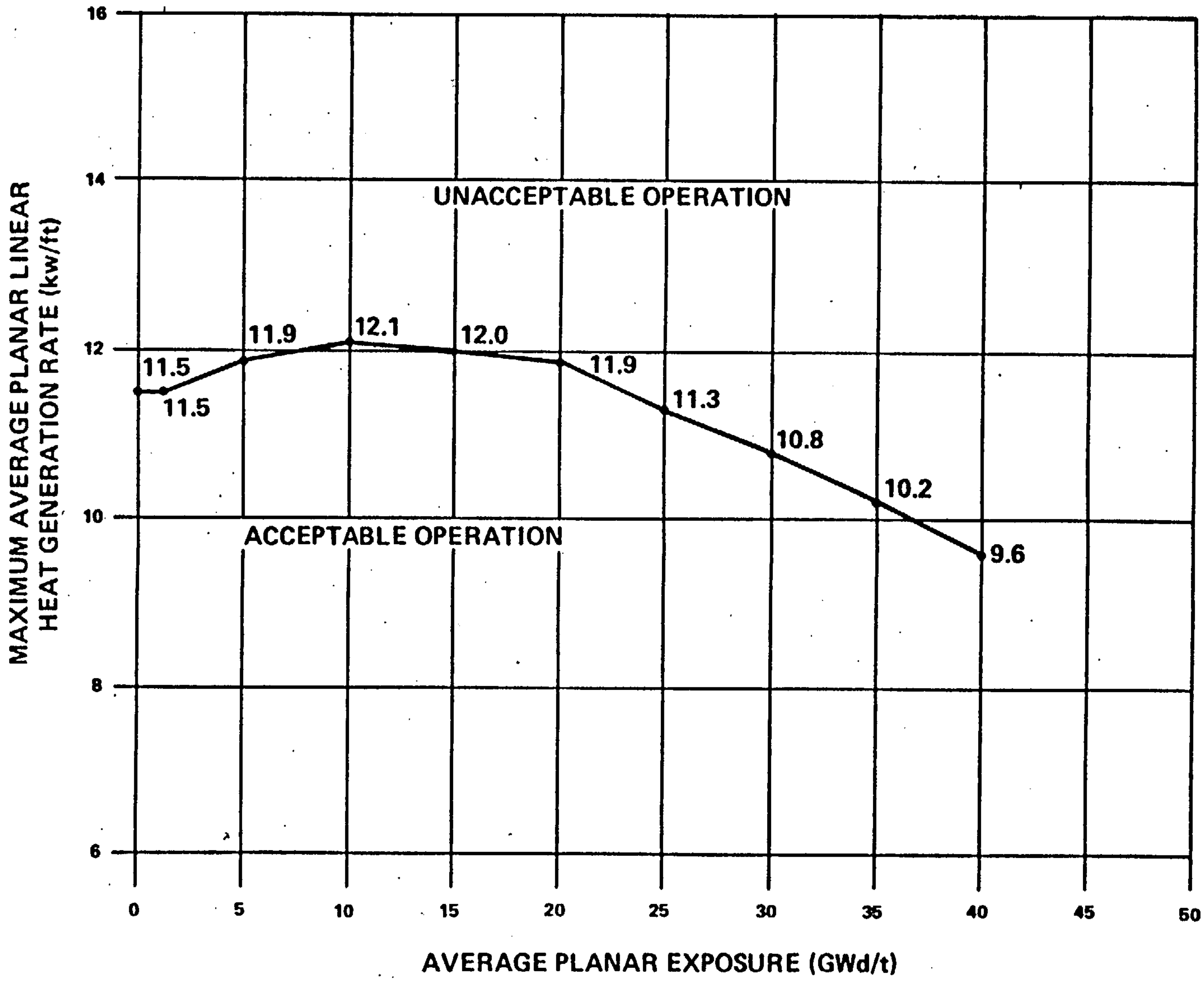
a. At least once per 24 hours,

b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

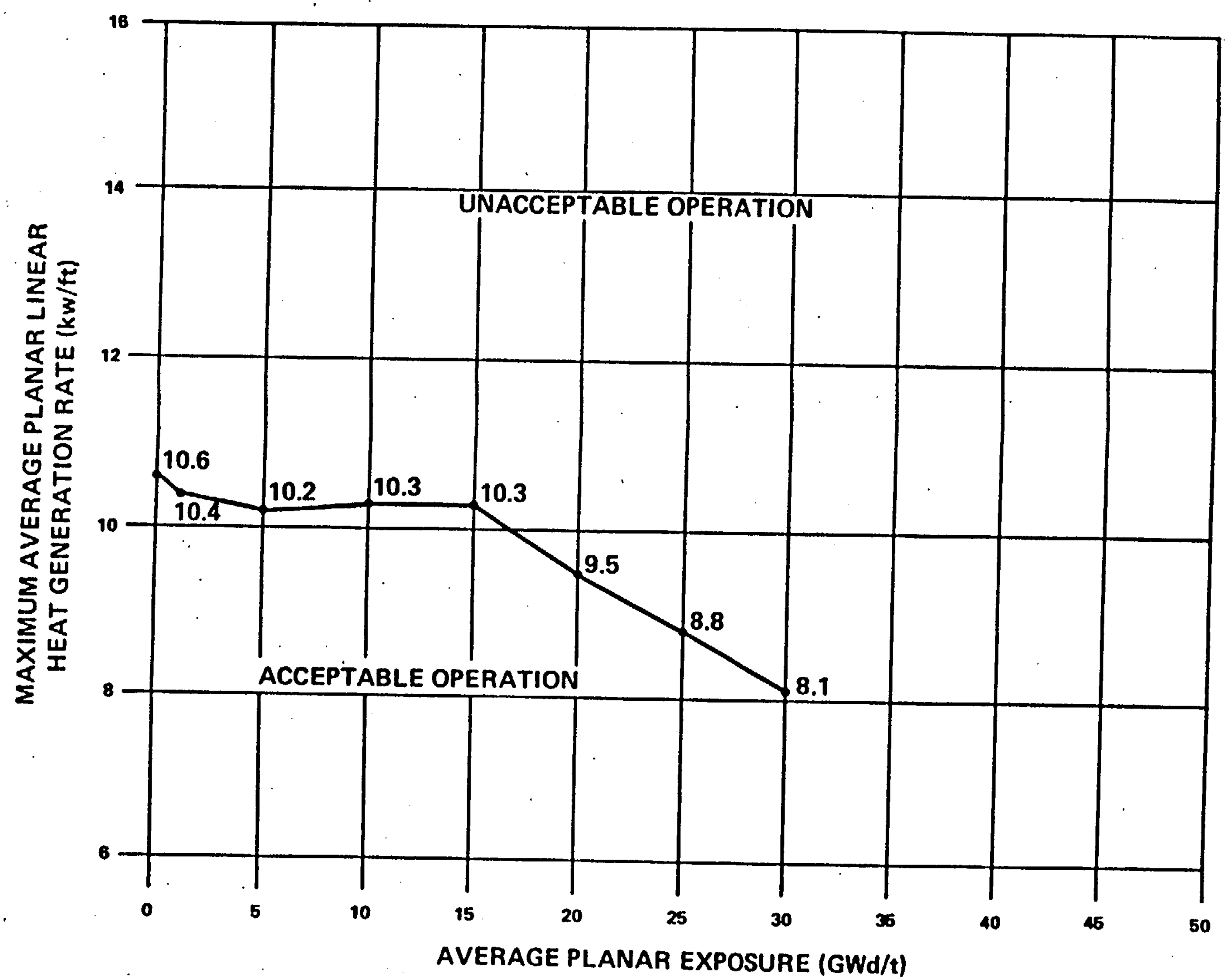
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.



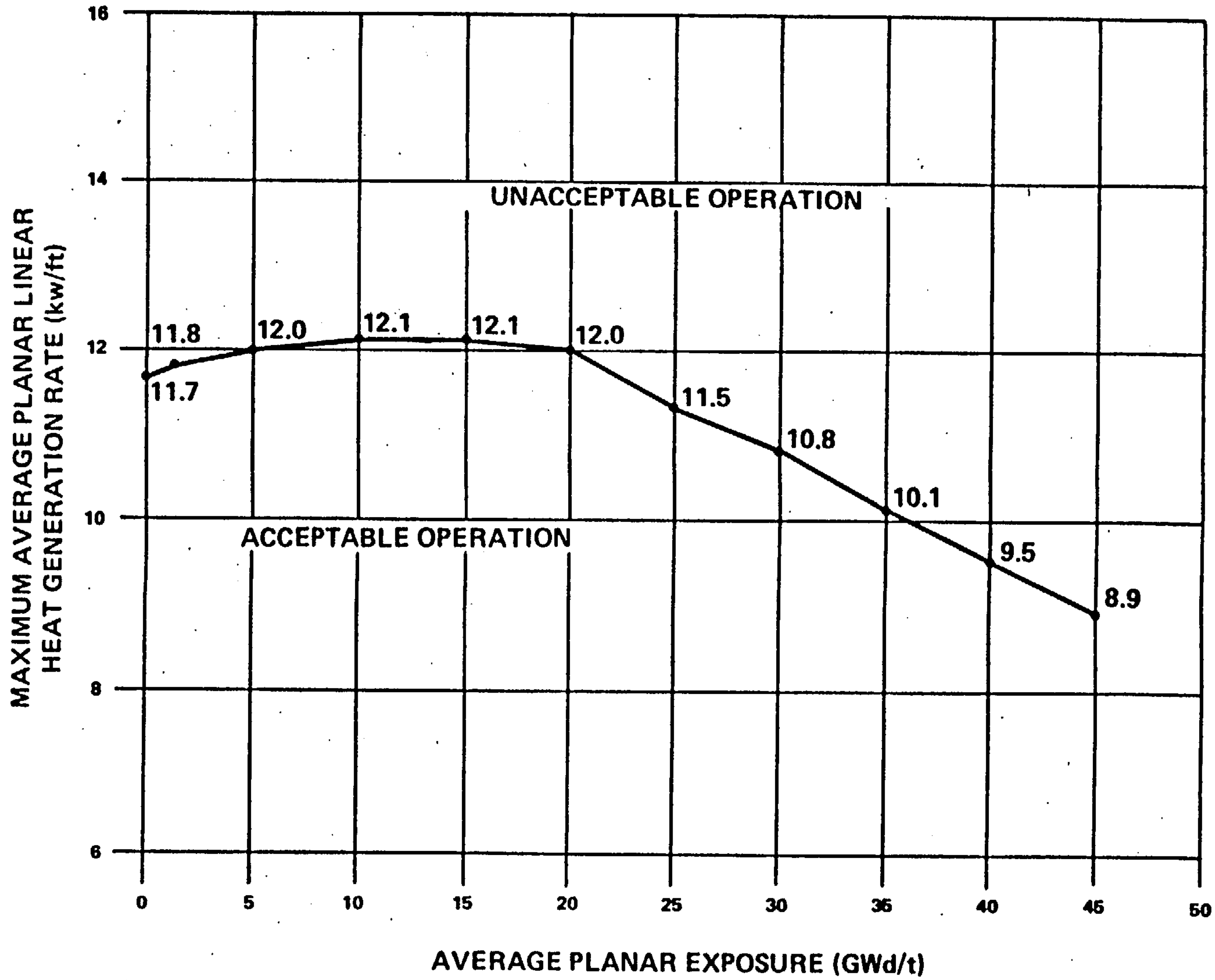
FUEL TYPE 8DIB175 (8DRL183) 100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-1



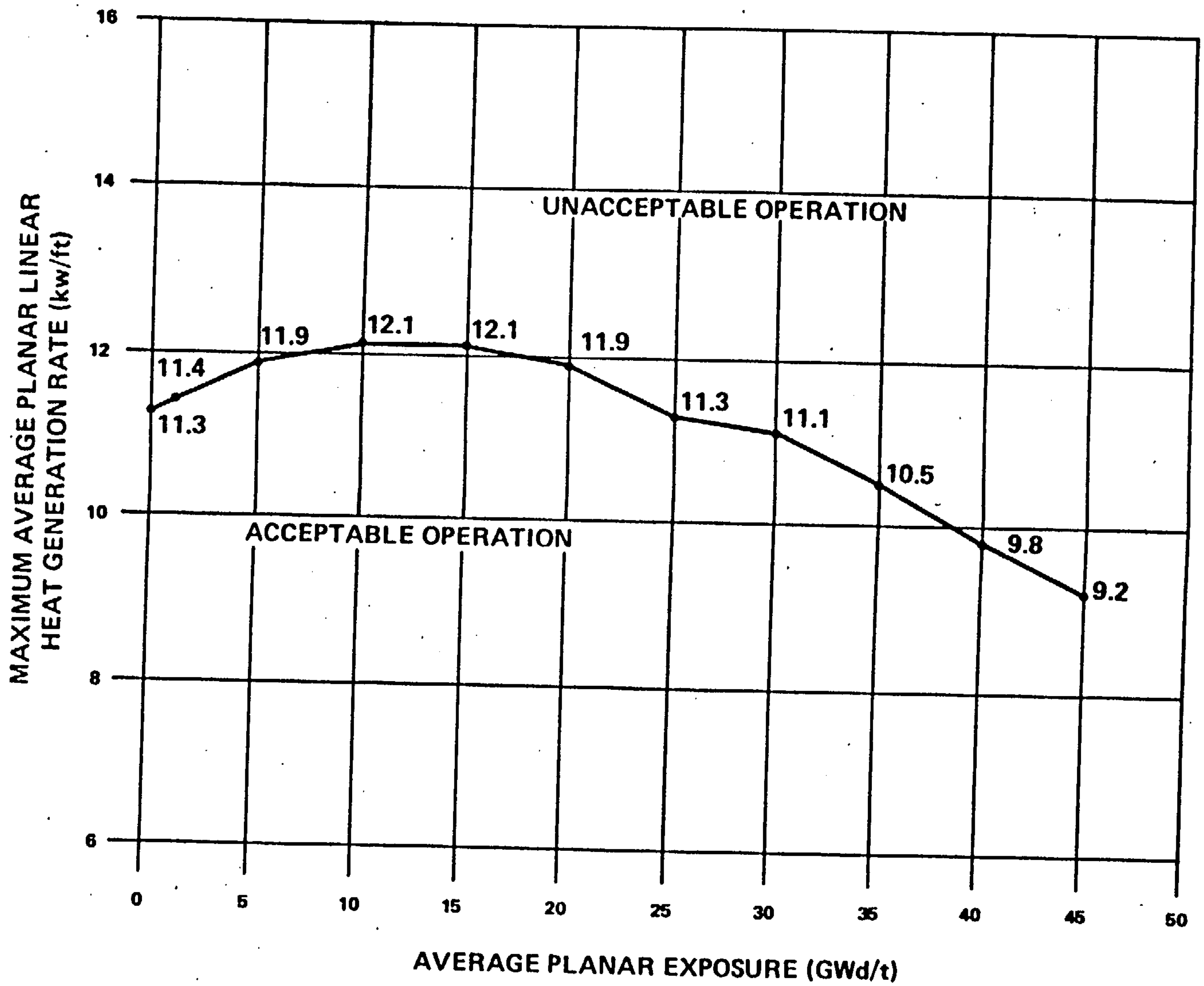
FUEL TYPE 8DIB221 (8DRL233) 100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2-1-2



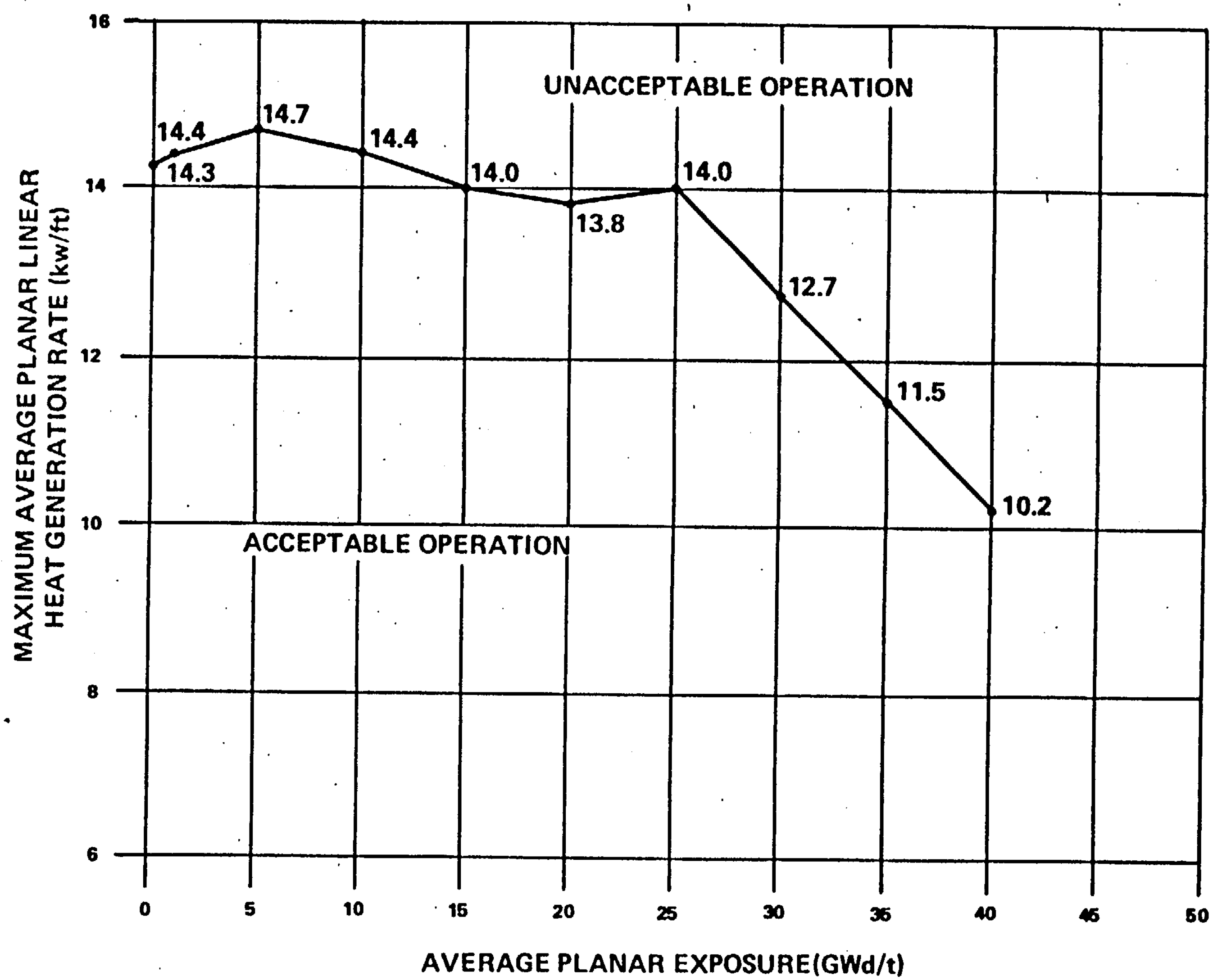
FUEL TYPE IE 711-00GD-100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-3



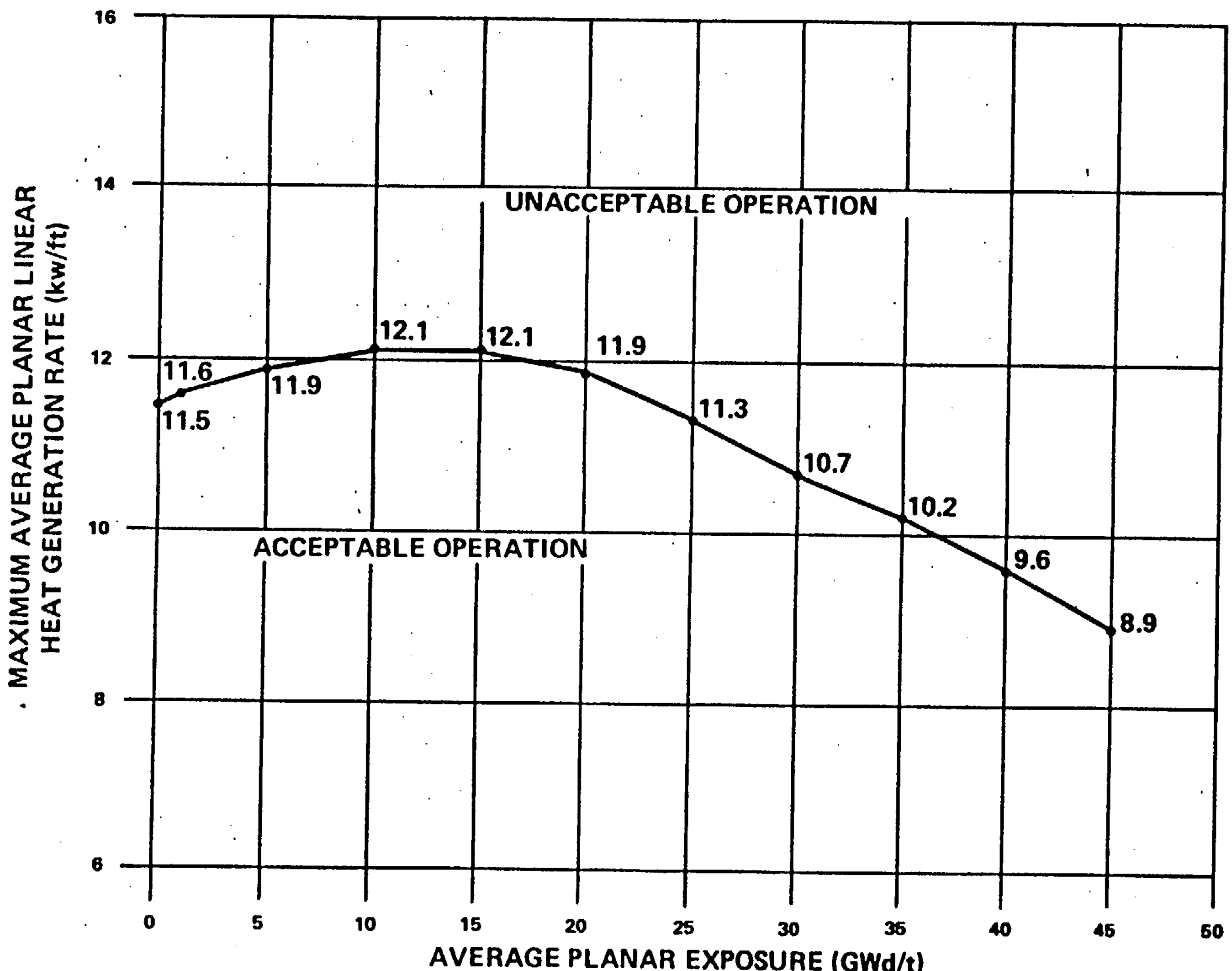
FUEL TYPES P8DRB284LA AND BP8DRB284LA 100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR EXPOSURE
FIGURE 3.2.1-4



FUEL TYPES P8DRB283 AND BP8DRB283 100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-5

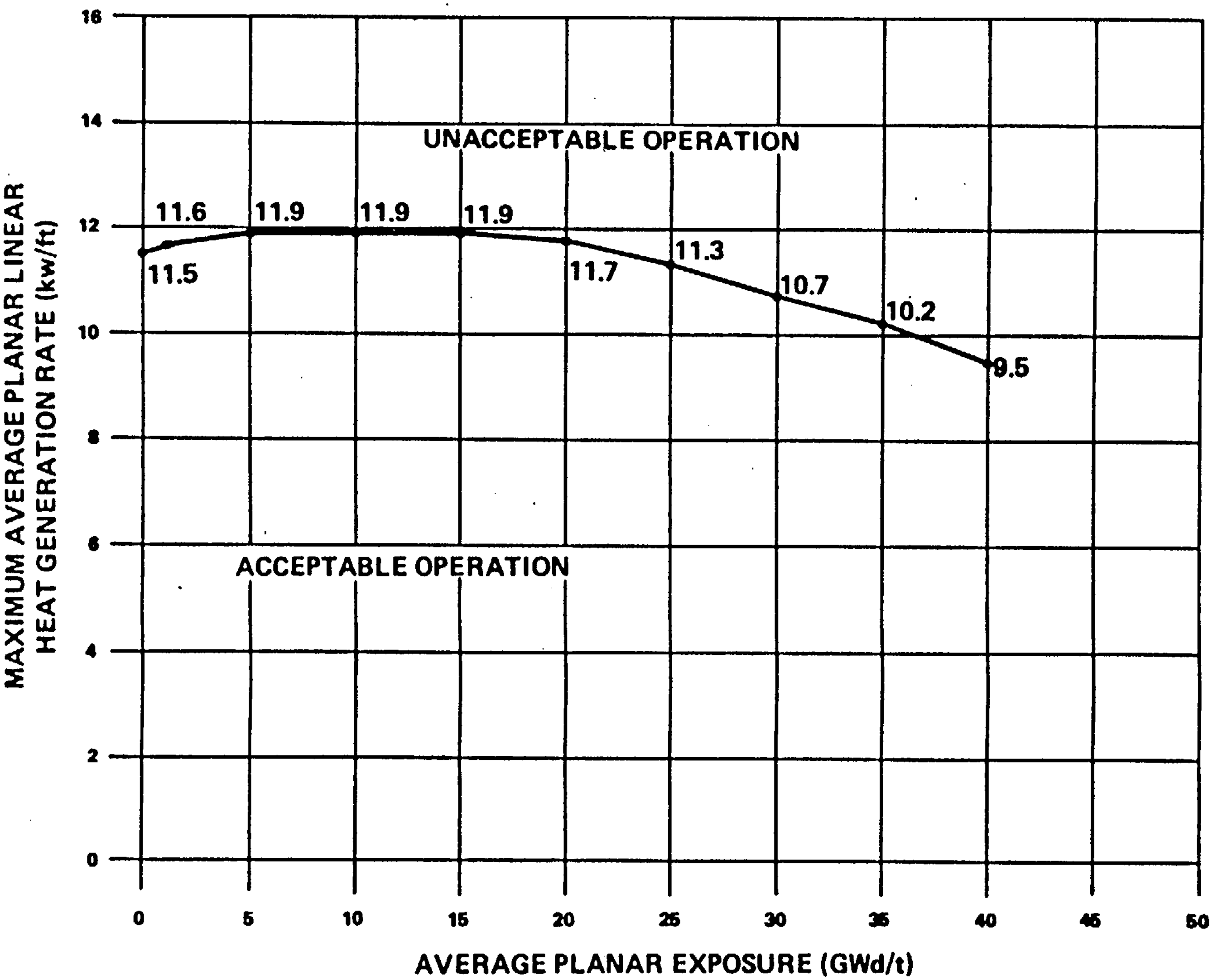


FUEL TYPES HATCH-1 I.C. 1,2,3 (7X7) 80 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MPALHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-6

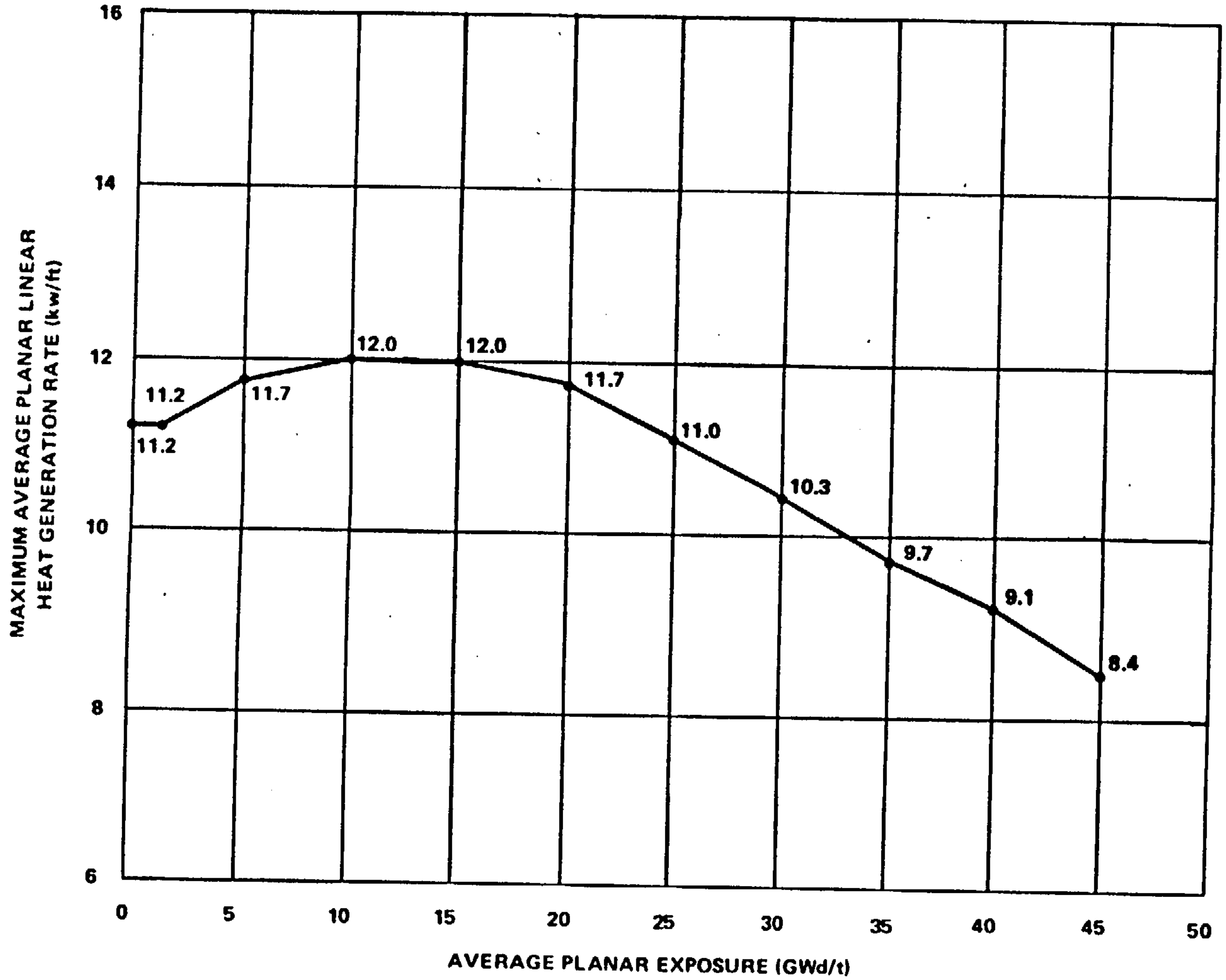


FUEL TYPES P8DRB265H AND BP8DRB265H 80 AND 100 MIL CHANNELS MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

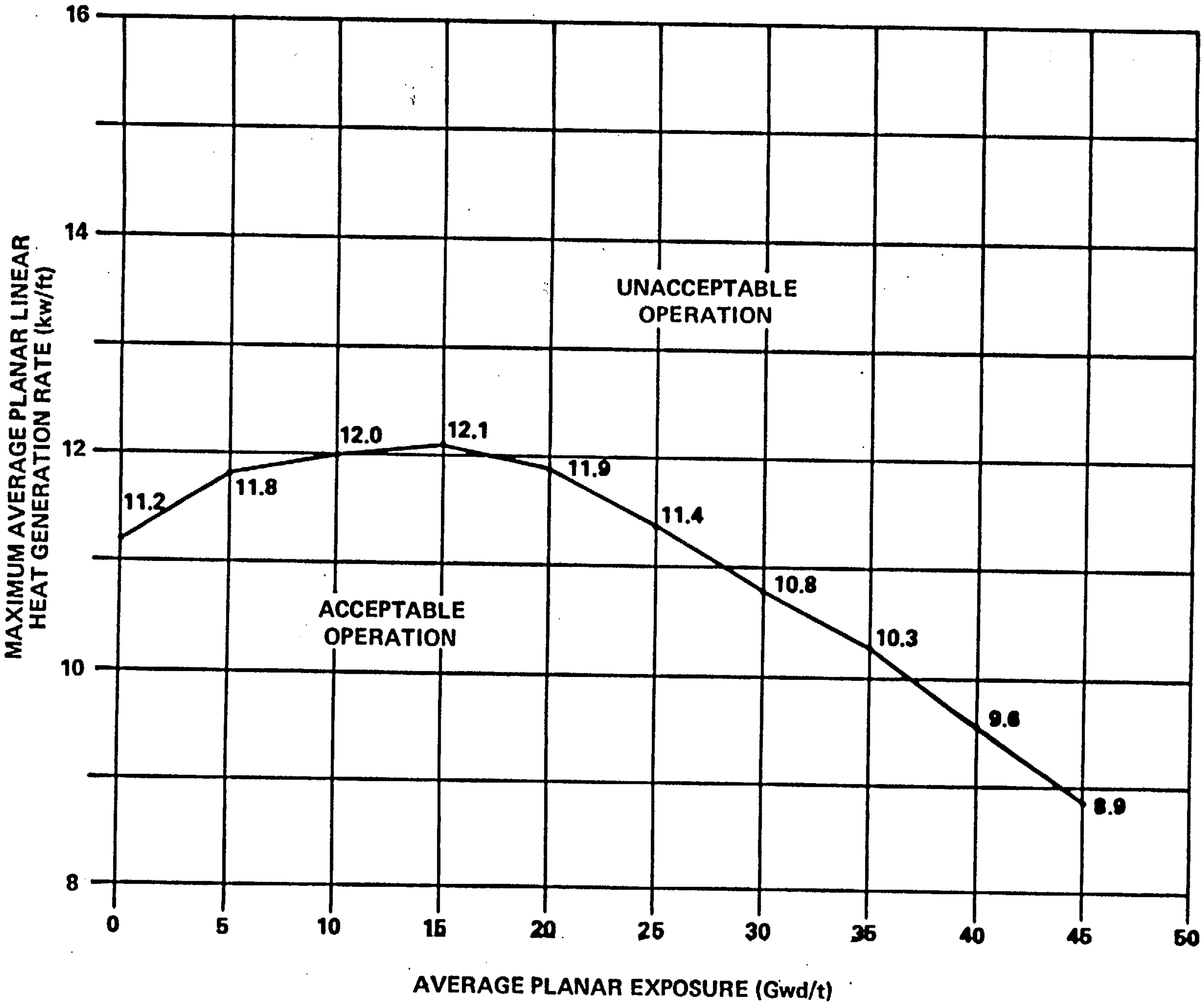
FIGURE 3.2.1-7



FUEL TYPE 8DRB265H 80 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-8

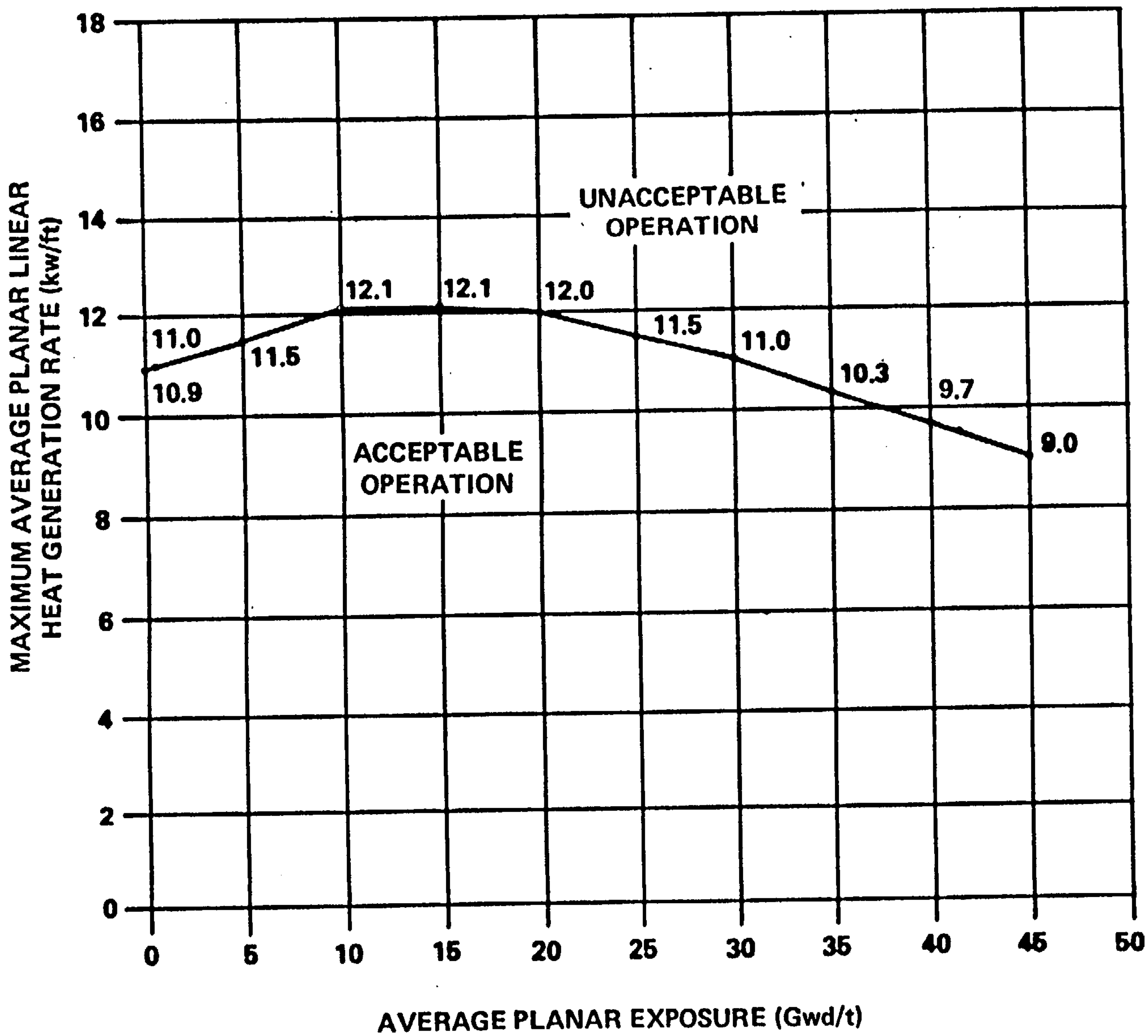


FUEL TYPES P8DRB284H AND BP8DRB284H 80 AND 100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-9

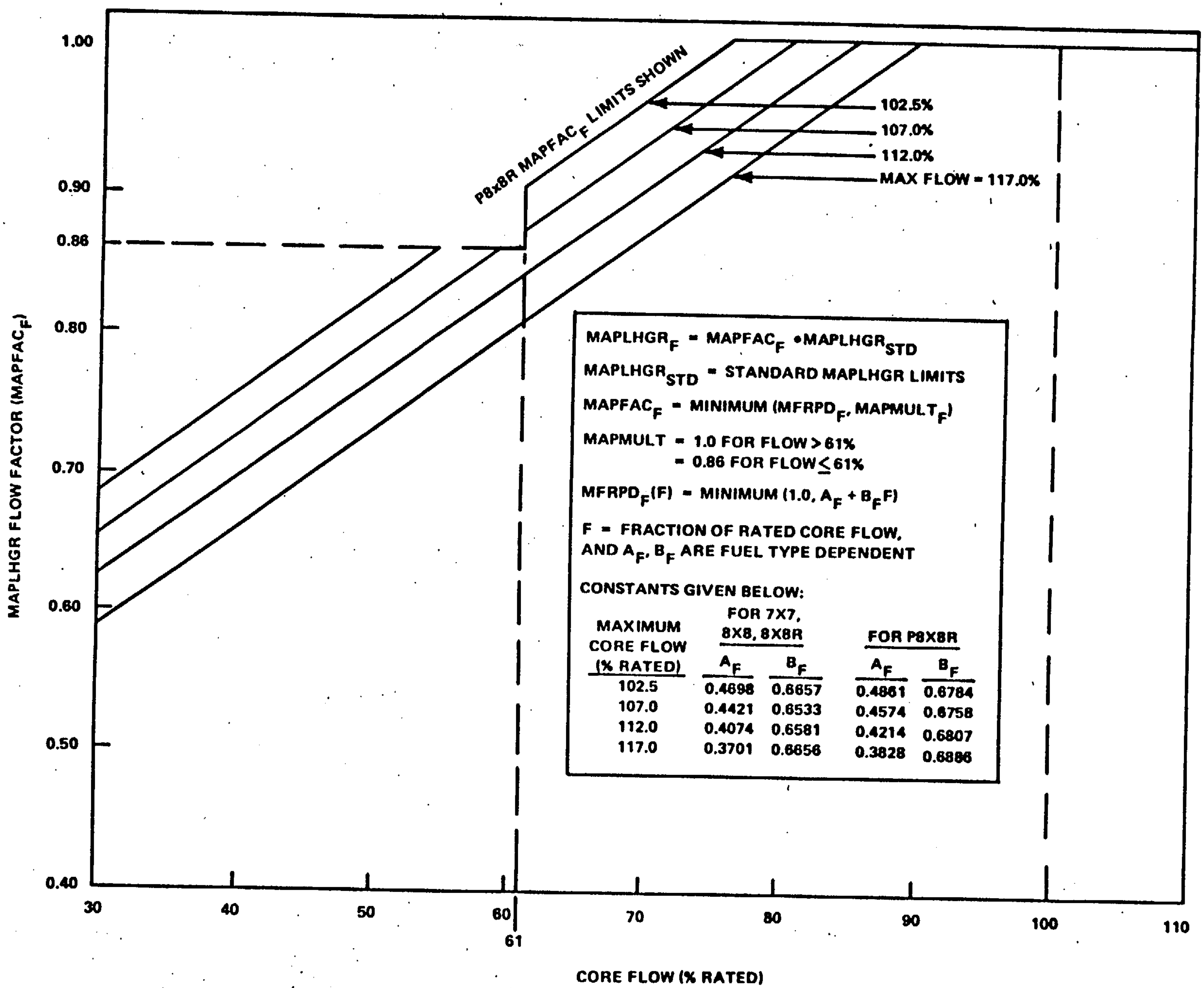


FUEL TYPES P8DRB283 AND BP8DRB283 80 MIL CHANNELS MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-10



FUEL TYPE BP8DRB299 80 AND 100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-11



$MAPLHGR_F = MAPFAC_F \cdot MAPLHGR_{STD}$

$MAPLHGR_{STD}$ = STANDARD MAPLHGR LIMITS

$MAPFAC_F$ = MINIMUM ($MFRPD_F$, $MAPMULT_F$)

MAPMULT = 1.0 FOR FLOW > 61%
= 0.86 FOR FLOW ≤ 61%

$MFRPD_F(F)$ = MINIMUM (1.0, $A_F + B_F F$)

F = FRACTION OF RATED CORE FLOW, AND A_F, B_F ARE FUEL TYPE DEPENDENT

CONSTANTS GIVEN BELOW:

MAXIMUM CORE FLOW (% RATED)	FOR 7X7, 8X8, 8X8R		FOR P8X8R	
	A_F	B_F	A_F	B_F
102.5	0.4698	0.6657	0.4861	0.6784
107.0	0.4421	0.6533	0.4574	0.6758
112.0	0.4074	0.6581	0.4214	0.6807
117.0	0.3701	0.6656	0.3828	0.6886

FIGURE 3.2.1-12 $MAPFAC_F$

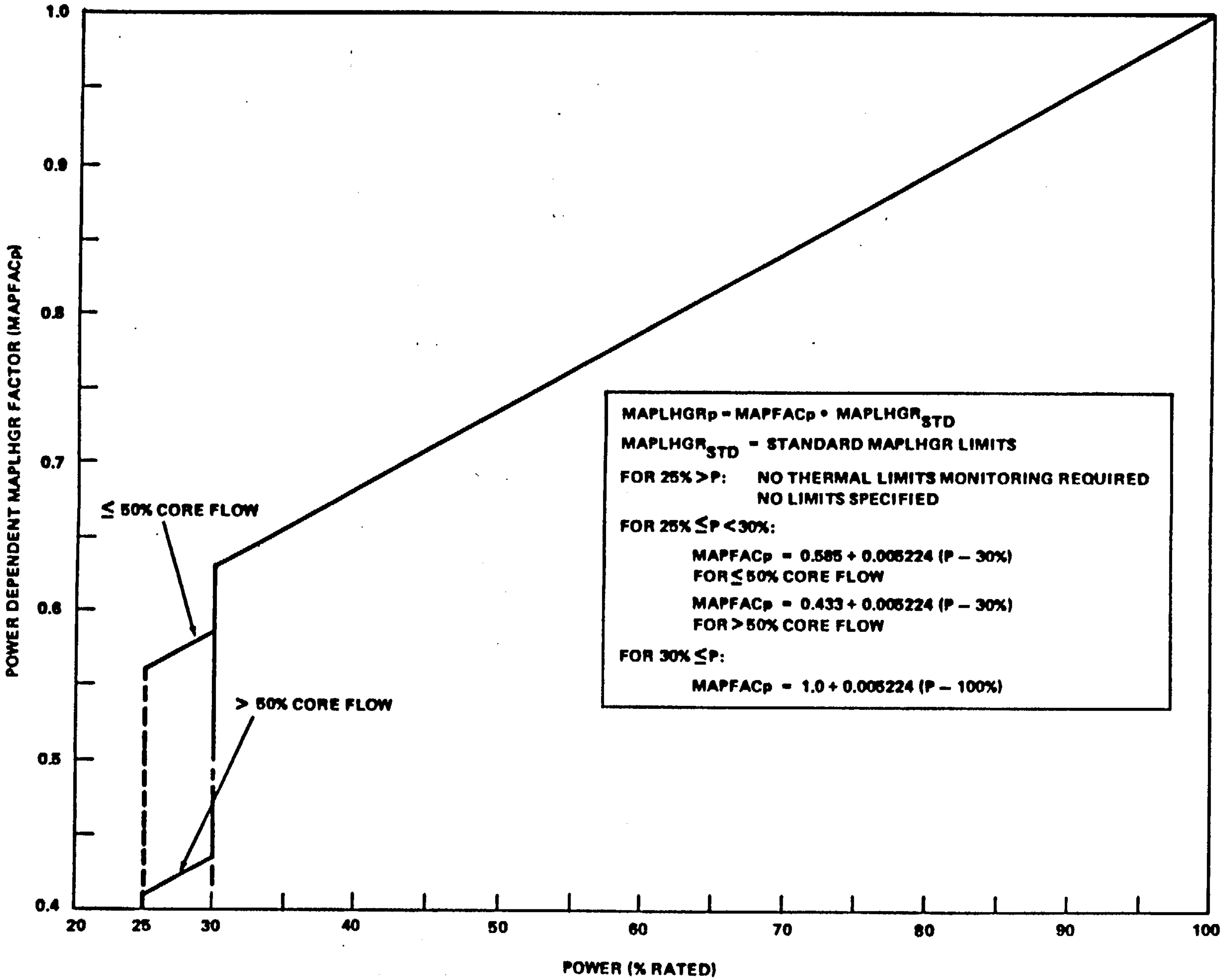


FIGURE 3.2.1-13 MAPFACp

3/4.2.2 APRM SETPOINTS

This section deleted.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 ALL MINIMUM CRITICAL POWER RATIOS (MCPRs), shall be equal to or greater than the MCPR operating limit (OLMCPR), which is a function of average scram time, core flow, and core power. For 25% ≤ Power < 30%, the OLMCPR is given in Figure 3.2.3-4. For Power ≥ 30%, the OLMCPR is the greater of either:

a. The applicable limit determined from Figure 3.2.3-3, or

b. The appropriate K_p given by Figure 3.2.3-4, multiplied by the appropriate limit from Figure 3.2.3-1 or 3.2.3-2 where:

$$\tau = 0 \text{ or } \left[\frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}\right], \text{ whichever is greater,}$$

τ_A = 1.096 sec (Specification 3.1.3.3 scram time limit to notch 36),

$$\tau_B = 0.834 + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2} (0.059),$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 36 of all rods measured in the i^{th} surveillance test, and

N_1 = total number of active rods measured in 4.1.3.2.a.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% RATED THERMAL POWER

ACTION:

With MCPR less than the applicable limit determined from Specification 3.2.3.a, or 3.2.3.b, initiate corrective action within 15 minutes and continue corrective action so that MCPR is equal to or greater than the applicable limit within 2 hours or reduce THERMAL POWER to less than or equal to 25% of RATED THERMAL POWER within the next 4 hours.

Amendment No. ~~21~~, ~~33~~, ~~39~~, 66

SURVEILLANCE REQUIREMENTS

4.2.3 The MCPR limit at rated flow and rated power shall be determined for each type of fuel (8X8R, P8X8R, BP8X8R, and 7X7) from Figures 3.2.3-1 and 3.2.3-2 using

a. τ = 1.0 prior to the initial scram time measurements for the cycle performed in accordance with Specification 4.1.3.2.a, or

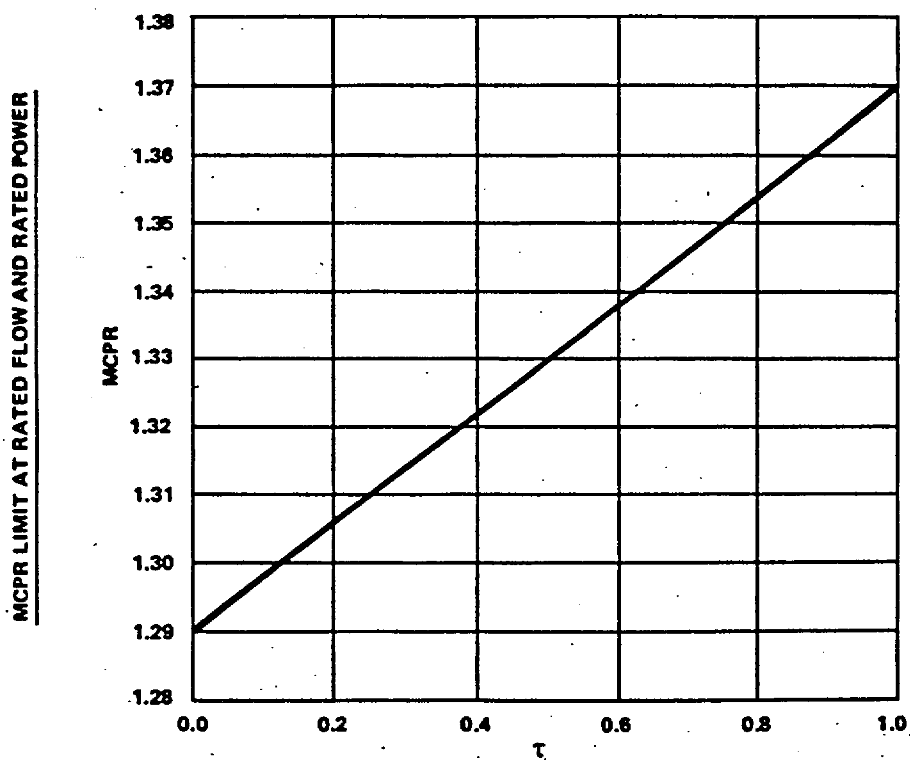
b. τ as defined in Specification 3.2.3; the determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2.

MCPR shall be determined to be equal to or greater than the applicable limit:

a. At least once per 24 hours,

b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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



ALL 8X8 FUEL TYPES
FIGURE 3.2.3-1

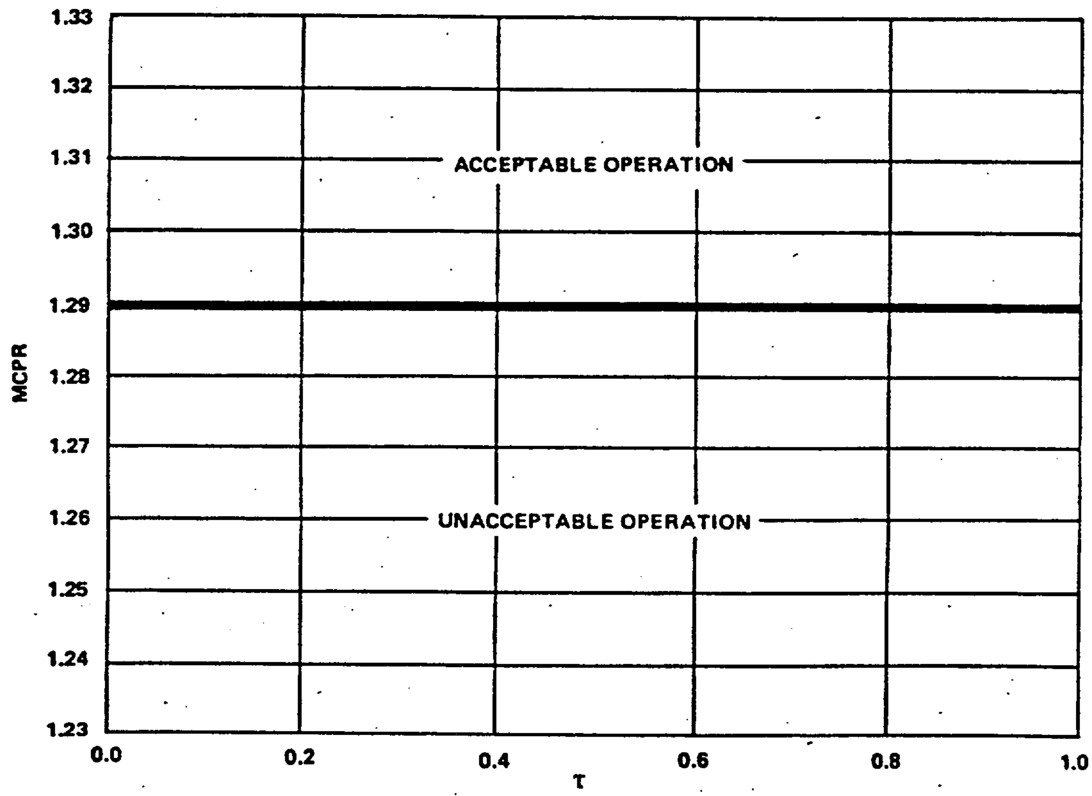


FIGURE 3.2.3-2
MCPR LIMIT FOR 7X7 FUEL
AT RATED FLOW AND RATED POWER

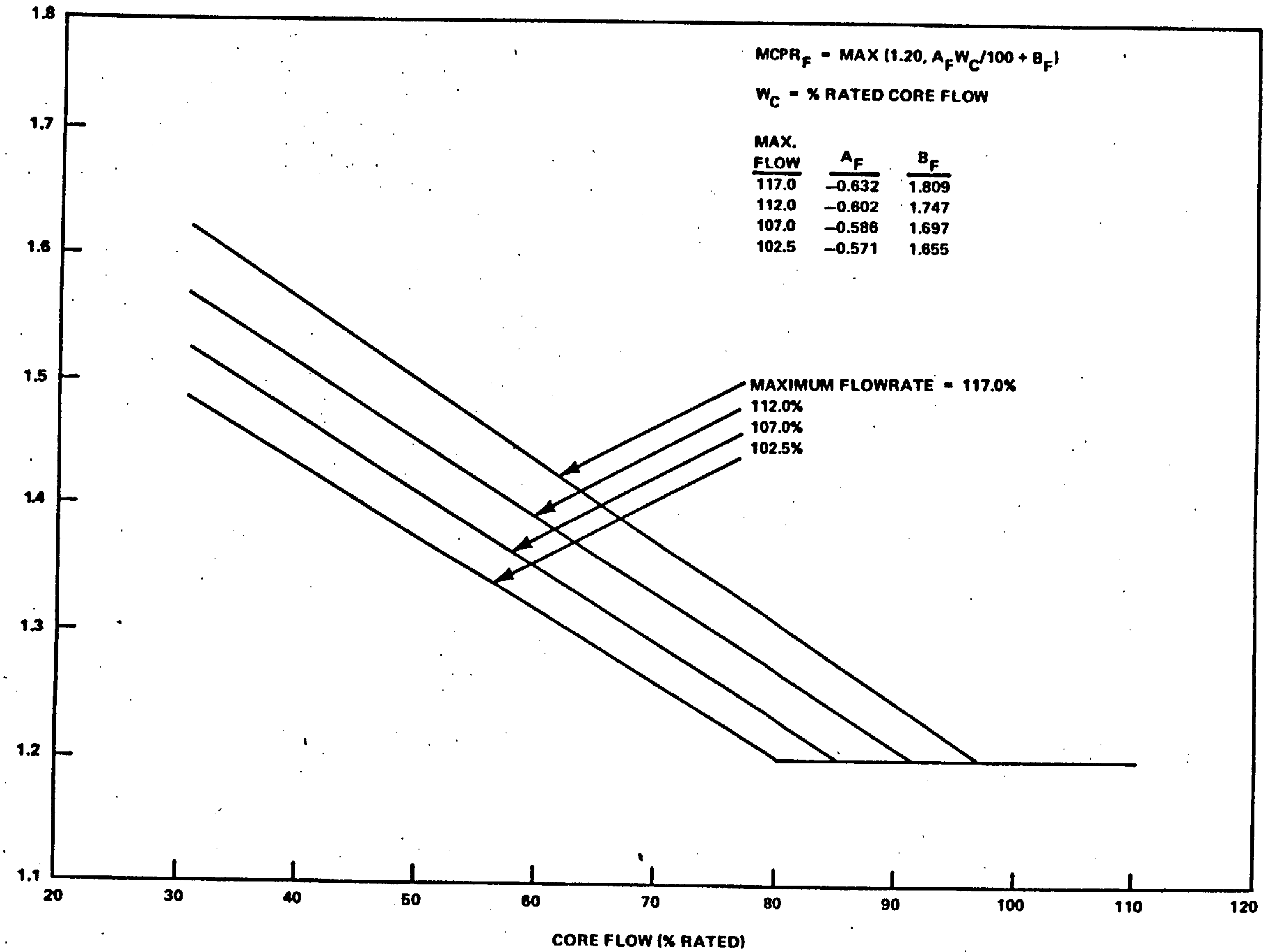


FIGURE 3.2.3-3 $MCPR_F$

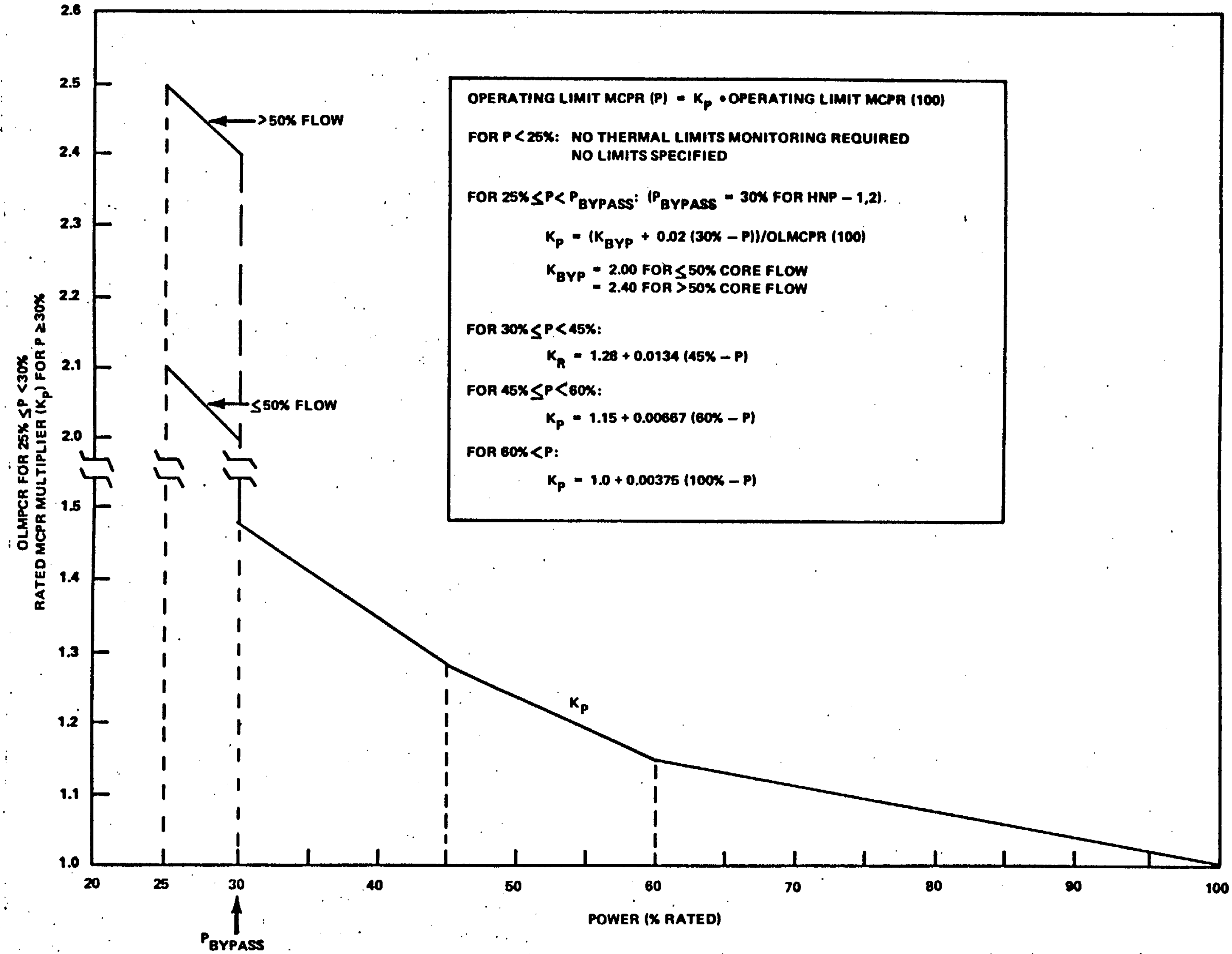


FIGURE 3.2.3-4 Kp

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 ALL LINEAR HEAT GENERATION RATES (LHGRs) shall not exceed 13.4 Kw/ft for 8X8R/P8X8R/BP8X8R fuel or 18.0 Kw/ft for 7X7 fuel.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and continue corrective action so that the LHGR is 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. When THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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

CONTROL RODS (Continued)

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 reactors.

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 each refueling. 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 cause the peak fuel enthalpy for any postulated control rod accident to exceed 280 cal/gm. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is ≥ 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 RWM to be OPERABLE below 20% of RATED THERMAL POWER and the RSCS to be OPERABLE from 50% control rod density to 20% of RATED THERMAL POWER provides adequate control.

CONTROL ROD PROGRAM CONTROLS (Continued)

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.1.38 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

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. The RBM is only required to be operable when the Limiting Condition defined in Specification 3.1.4.3 exists. 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. Further discussion of the RBM system and power dependent setpoints may be found in NEDC-30474-P (Ref. 4).

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for maintaining the reactor subcritical in the event that insufficient rods are inserted in the core when a scram is called for. The volume of the poison solution and weight percent of poison material in solution is based on being able to bring the reactor to the subcritical condition as the plant cools to ambient condition. The temperature requirement is necessary to keep the sodium pentaborate in solution. Checking the volume and temperature once each 24 hours assures that the solution is available for use.

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 water is added; thus, a check on the temperature and volume once each 24 hours assures that the solution is available for use.

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 the Final Acceptance Criteria (FAC) issued in June 1971 considering the postulated effects of fuel pellet densification. These specifications also assure that fuel design margins are maintained during abnormal transients.

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, Appendix K.

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 an 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 APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in the figures for in Technical Specification 3/4.2.1.

The calculational procedure used to establish the APLHGR shown in the figures in Technical Specification 3/4.2.1 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses performed with Reference 1 are: (1) the analysis assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in the figures in Technical Specification 3/4.2.1; (2) fission product decay is computed assuming an energy release rate of 200 MEV/fission; (3) pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; and (4) the effects of core spray entrainment and counter-current flow limitation as described in Reference 2, are included in the reflooding calculations.

A flow dependent correction factor incorporated into Figure 3.2.1-12 is applied to the rated conditions APLHGR to assure that the 2200 F°PCT limit is complied with during a LOCA initiated from less than rated core flow. In addition, other power and flow dependent corrections given in Figures 3.2.1-12 and 3.2.1-13 are applied to the rated conditions to assure that the fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from off-rated conditions.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in bases Table B 3.2.1-1. Further discussion of the APLHGR limits is given in Reference 4.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

FOR HATCH-UNIT 2

Plant Parameters:

Core Thermal Power .... 2531 Mwt which corresponds to 105% of license core power*

Vessel Steam Output .... 10.96×10^6 lbm/h which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure .... 1055 psia

Design Basis Recirculation Line Break Area For:

a. Large Breaks .... 4.0, 2.4, 2.0, 2.1 and 1.0 ft²

b. Small Breaks .... 1.0, 0.9, 0.4 and 0.07 ft²

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.18

A more detailed list of input to each model and its source is presented in Section II of Reference 1 and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification linear heat generation rate limit.

3/4.2.2 APRM SETPOINTS

This section deleted.

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 of 1.07, 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 as given in Specification 2.2.1.

To assure that the fuel cladding integrity Safety Limits are not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which results 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.

MINIMUM CRITICAL POWER RATIO (Continued)

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.1-6 that are input to a GE-core dynamic behavior transient computer program described in NEDO-10802[3]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDO-20566[1]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$, and the Kp of Figures 3.2.3-3 and 3.2.3-4, respectively 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 s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam 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 the 105% of rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

The core power dependent MCPR operating limit MCPR is the power rated flow MCPR operating limit multiplied by the K_p factor given in Figure 3.2.3-4.

The Kps are established to protect the core from transients other than core flow increases, including the localized event such as rod withdrawal error. The K_ps were determined based upon the most limiting transient at the given core power level. For further information on MCPR operating limits for off-rated conditions, reference NEDC-30474-P.[4]

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the refuel position ensures that the restrictions on rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage the reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. During the unloading, it is not necessary to maintain 3 cps because core alterations will involve only reactivity removal and will not result in criticality. The loading of up to four bundles around the SRMs before attaining the 3 cps is permissible because these bundles were in subcritical configuration when they were removed and therefore will remain subcritical when placed back in the previous positions.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod and prevents two positive reactivity changes from occurring simultaneously.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building provides secondary containment during normal operation when the drywell is sealed and in service. When the reactor is shutdown or during refueling, the drywell may be open and the reactor building then becomes the primary containment. The refueling floor is maintained under the secondary containment integrity of Hatch-Unit 1.

Establishing and maintaining a vacuum in the building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches and dampers, is adequate to ensure that there are no violations of the integrity of the secondary containment. Only one closed damper in each penetration line is required to maintain the integrity of the secondary containment.

3/4.9.6 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.7 CRANE AND HOIST OPERABILITY

The OPERABILITY requirements of the cranes and hoists used for movement of fuel assemblies ensures that: (1) each has sufficient load capacity to lift a fuel element, and (2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.8 CRANE TRAVEL-SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel element over irradiated fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analyses. All fuel loaded into the Edwin I. Hatch Nuclear Plant spent fuel pool shall have an uncontrolled lattice k_{∞} less than or equal to the limit for high density fuel racks described in the "General Electric Standard Application for Reactor Fuel" (GESTAR II), NEDE-24011-P-A-8. Alternatively, fuel not described in GESTAR II shall have been analyzed with another NRC approved methodology to ensure conformity to the FSAR design basis for fuel in the spent fuel racks.

3/4.9.9 and 3/4.9.10 WATER LEVEL-REACTOR VESSEL AND WATER LEVEL-SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 CONTROL ROD REMOVAL

This specification ensures that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone coincides with the exclusion area and is also shown in Figure 5.1.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel structure composed of a series of vertical right cylinders and truncated cones which form a drywell. This drywell is attached to a suppression chamber through a series of vents. The suppression chamber is a steel pressure vessel in the shape of a torus. The primary containment has a total minimum free air volume of 255,978 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum design internal pressure 56 psig.

b. Maximum allowable internal pressure 62 psig.

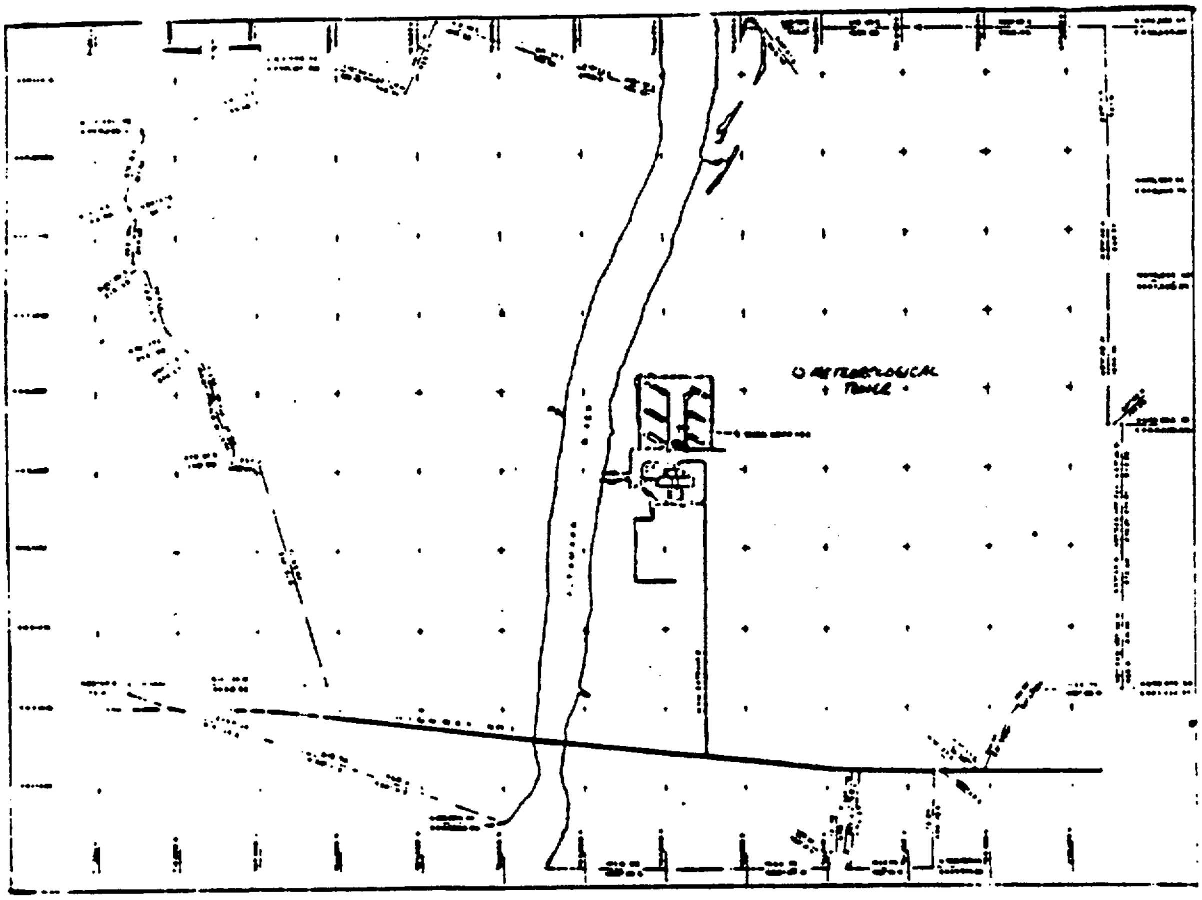
c. Maximum internal temperature 340°F.

d. Maximum external pressure 2 psig.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall consist of not more than 560 fuel assemblies 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 Final Safety Analysis Report (FSAR).



EXCLUSION AREA AND LOW POPULATION ZONE

FIGURE 5.1.1-1

DESIGN FEATURES

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 137 cruciform-shaped control rod assemblies.

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 1250 psig, and

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 17,050 cubic feet at a nominal T_{ave} of 540°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to ≤ 0.95 when flooded with unborated water. The k_{eff} of ≤ 0.95 includes conservative allowances for uncertainties.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 185 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2845 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 132 AND 66 TO

FACILITY OPERATING LICENSES NOS. DPR-57 AND NPF-5

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNITS NOS. 1 AND 2

DOCKETS NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated April 15, 1986 (Reference 1) Georgia Power Company (GPC) proposed to change the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant Units 1 and 2 in four areas. These areas are related to (1) requirements for the Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS), (2) restrictions on the fuel to be stored in the spent fuel pools, (3) additions to the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit curves for several new fuel assemblies, (4) editorial changes to correct editorial errors and to clarify applicability of limits to various fuel assembly types. The initial submittal was supplemented by additional information (Reference 2) in response to staff questions. These requests are further described as follows.

1. It is proposed that the use of Technical Specification requirements for RWM and RSCS be modified to permit the use of Banked Position Withdrawal Sequences (BPWS) for the first 50 percent of control rod withdrawal. This modified pattern would be enforced by the RWM alone in this regime. The RWM and RSCS would continue to enforce the Group Notch pattern above 50 percent rod withdrawal as in the past. This mode of operation has been discussed in Amendment 12 to GESTAR II (Reference 3) and in the staff Safety Evaluation Report on this subject (Reference 4). Its chief advantage to the utility is that with BPWS a limiting generic analysis may be used for the Rod Drop Accident (RDA) analysis

and a reactor-cycle specific analysis is not needed. From an NRC viewpoint it provides smaller rod reactivity worths should a RDA event occur.

In Reference 2 GPC responded to staff questions on maximal use of the RWM in the BPWS withdrawal range and to the quality control aspects of the use of second operators when the RWM is not operational. It is noted that the BPWS has been procedurally implemented for several years at Hatch (without RDA credit) and the experience with the RWM has resulted in high reliability and minimum bypass. When required, the second operator procedures and quality control has been substantially examined and improved over the past several years as a result of a previous enforcement action relating to control rod operation.

2. GPC proposes to remove from Section 5 of the TS relating to the spent fuel pool the fuel assembly linear mass restrictions of 15.2 grams of Uranium-235 per centimeter. Also proposed is the removal from Section 5 of other descriptions of mechanical features of the fuel assemblies that may be in the core.

General Electric (GE) now provides in GESTAR II (Reference 5) a staff approved approach to fuel assembly limits in the spent fuel pool based on k-infinity of the assemblies, without specific regard to linear mass limits. GESTAR II provides specific k-infinity limits for assemblies (of GE design) to be stored in spent fuel racks of GE design in order to maintain a (staff approved) fuel pool multiplication limit of less than 0.95, including uncertainties. Maximum k-infinities for each GE fuel design are also given. Hatch spent fuel racks are GE designed high-density racks and as such have a k-infinity limit of 1.33 for GE fuel according to Amendment 13 to GESTAR II (Reference 6). Current Hatch fuel assemblies are all of GE design.

In addition to the changes relating to linear mass requirements, changes are also proposed for descriptions of fuel assemblies in the core which would eliminate specific mechanical descriptions (e.g., 62 rods) and refer instead to assemblies analyzed with NRC approved methods and which comply with Design Bases in the Hatch Final Safety Analysis Report (FSAR).

3. GPC proposed to add new fuel types and channel thickness to its inventory. This requires additional MAPLHGR curves for the TS. These fuel assemblies have been analyzed for the Loss-of-Coolant Accident event and resulting MAPLHGR limits for Hatch by GE using NRC approved methods (see Attachment 2 to Reference 1). The results are applicable to both Hatch 1 and 2. The MAPLHGR results are applicable to assemblies P8DRB283, BP8DRB283 and BP8DRB299 with 80 mil channel thicknesses which are added to the Hatch 1 and 2 TS.

4. GPC proposed two editorial changes for Hatch 2 to correct a typographical error in Basis Table 3.2.1-1 and a title change to (new) Figure 3.2.1-12. There are also editorial changes to MAPLHGR figures for both units to specify which thermal limits apply to which channel thickness and to barrier or nonbarrier fuel. There are also some figure number changes because of the added MAPLHGR curves.

2.0 EVALUATION

1. A previous generic review (Reference 4) has concluded that the use of a RWM enforced BPWS pattern for the first 50 percent of rod withdrawal in a Group Notch RSCS plant is acceptable and in fact preferable, and plants making the change will be able to take credit for the statistical analysis of the RDA and will not have to analyze the event for reloads. This approval is applicable to Hatch 1 and 2 upon adoption of the BPWS procedures.

   To permit the procedures, TS changes are made to Specifications 3/4.3.G (and Basis 3/4.3.G) for Hatch 1 and to 3/4.1.4.1 and .2 for Hatch 2. These changes appropriately address RWM enforced BPWS patterns for the first 50 percent and RWM and RSCS enforced Group Notch beyond 50 percent withdrawal. They are acceptable.

GPC response to questions on RWM operability and second operator effectiveness indicate that, as a result of previous implementation of BPWS patterns and a previous problem with rod movement error resulting in Enforcement Action, particular special attention has been given to these areas for several years. This attention appears to have resulted in appropriate procedure and quality control examination and improvement in these areas, and the response is acceptable.

2. The removal of previous requirements for a U235 linear mass limit for the spent fuel pool has become generally accepted practice. This change is based on the staff approval of the GE approach (described in GESTAR II) of determining, for a given fuel pool design, limits on the k-infinity of fuel assemblies which may be placed in the pool racks, and providing for each GE fuel assembly design a maximum k-infinity which can exist over the fuel burnup range. This change from the U235 content limit results from the recognition of the reactivity effect of the gadolinium burnable poison in the fuel assembly. The basic requirement for the fuel pool, that the neutron multiplication remain less than 0.95, remains unchanged. For Hatch the (Section 5) TS limit of 0.95 remains and the corresponding basis will refer to the GESTAR II description of the approach and limits. Current and presently planned future Hatch fuel is described in the current GESTAR II (Revision 7) and the (approved) assembly k-infinity values are within the limit for the GE designed Hatch high density racks. The proposed approach and TS change is thus acceptable for Hatch 1 and 2, when using GE fuel. The methodology for determining k-infinity of assemblies by other fuel vendors would require further review.

The proposed change to the TS Section 5 description of mechanical features of fuel assemblies in the core eliminates a few not very useful descriptive terms and provides instead a requirement to use assemblies which have been analyzed with NRC approved methods and which comply with Safety Design Bases given in the Hatch FSAR. This is a reasonable change and is acceptable.

3. The new MAPLHGR curves for the new fuel assembly or new channel thickness proposed to be added to the Hatch 1 and 2 inventories have been calculated with NRC approved standard methodology by GE for Hatch 2, and since Hatch 2 results are conservative for Hatch 1 (Attachment 2 to Reference 1), they are also applicable for Hatch 1. These additions to the Hatch TS are therefore acceptable. They will be Figures 3.11.1 Sheets 5 and 6 for Hatch 1 and 3.2.1-10 and 11 for Hatch 2.

4. The editorial changes to correct a typographical error, retitle a figure, change page number and specify channel thickness and barrier fuel more clearly are all straightforward and acceptable.

We have reviewed the report and supplemental information submitted by GPC for proposed TS changes relating to BPWS operation, spent fuel pool limits, new MAPLHGR curves and editorial alterations. Based on this review we have concluded that appropriate material was submitted and that the proposed changes satisfy staff positions and requirements in these areas. Operations in the proposed manner and the TS changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

An Environmental Assessment has been prepared pursuant to 10 CFR Part 51.32 and published in the Federal Register on .

4.0 CONCLUSION

On the basis of the considerations discussed above, the staff has concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations.

Principal Contributor: H. Richings

Dated: October 31, 1986

REFERENCES

1. Letter (and Enclosures) from J. T. Beckham, Georgia Power Company (GPC), to D. Muller, NRC, dated April 15, 1986, "Edwin I. Hatch Nuclear Plant Units 1 and 2 Technical Specifications Revisions ...."

2. Letter from L. T. Gucwa, GPC, to D. Muller, NRC, dated July 25, 1986, ".... Supplemental Information."

3. Letter (and Enclosure) from J. Charnley, GE, to C. Thomas, NRC, Dated May 17, 1985, "Proposed Amendment 12 to GE Licensing Topical Report NEDE-24011-P-A."

4. Letter from C. Thomas, NRC, to J. Charnley, GE, dated October 11, 1985, "Acceptance for Referencing of .... NEDE-24011-P-A, ... Revision 6, Amendment 12."

5. NEDE-24011-P-A (GESTAR II), Revision 7, August 1985, "General Electric Standard Application for Reactor Fuel."

6. NEDE-24011-P-A (GESTAR II), Revision 6, Amendment 13.

UNITED STATES NUCLEAR REGULATORY COMMISSION

ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT

REGARDING PROPOSED AMENDMENTS TO FACILITY OPERATING LICENSES

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA
DOCKETS NOS. 50-321 AND 50-366

The U.S. Nuclear Regulatory Commission (the Commission) is considering the issuance of amendments to Facility Operating Licenses Nos. DPR-57 and NPF-5 issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia (the licensees) for operation of the Edwin I. Hatch Nuclear Plant, Units Nos. 1 and 2 (the facility) located in Appling County, Georgia.

ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action: The proposed action would permit the licensees to implement changes to Hatch Plant, Units 1 and 2, and Technical Specifications as described in their letter of April 15, 1986, as supplemented July 25 and September 22, 1986. The following assessment applies to Units 1 and 2.

The Need for the Proposed Action: The need for the proposed action is to:

(i) eliminate the need to analyze the control drop accident for each fuel cycle;

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(ii) increase the availability for power production and increase the overall efficiency; and

(iii) make editorial corrections.

Environmental Impacts of the Proposed Action: The proposed action would: a) permit use of Banked Position Withdrawal Sequences for the first 50 percent of control rod withdrawal, b) remove the linear mass restriction of 15.2 grams of Uranium-235 per centimeter for fuel assemblies stored in the fuel pool, c) eliminate specific mechanical descriptions of fuel assemblies, d) provide Maximum Average Planor Linear Heat Generation limit curves for several new fuel assemblies, and (e) make several editorial changes. The net power level is unchanged. The response of the reactor protection system under accident conditions is unchanged. Thus, post-accident radiological releases will not be greater than previously determined, nor does the proposed change otherwise affect radiological plant effluents. Occupational exposures to radiation would also be unaffected. Therefore, the Commission concludes that there are no significant radiological environmental impacts associated with the proposed amendments.

With regard to potential nonradiological impacts, the proposed change involves systems located within the restricted area as defined in 10 CFR 20. No nonradiological effluents are affected, and no other environmental impact would occur. Therefore, the Commission concludes that there are no significant nonradiological environmental impacts associated with the proposed change.

Since we have concluded that there is no measurable environmental impact associated with the proposed changes to the TSs, any alternatives to these changes will have either no environmental impact or greater environmental impact.

The principal alternative would be to deny the requested amendment. This would not reduce environmental impacts of plant operation.

Alternative Use of Resources: This action does not involve the use of resources not previously considered in connection with the Final Environmental Statements related to Operation of Hatch Unit 1 (Final Environmental Statement dated October 25, 1972) and Hatch Unit 2 (Final Environmental Statement dated March 1978).

Agencies and Persons Consulted: The Commission's staff reviewed the licensees' request and did not consult other agencies or persons.

FINDING OF NO SIGNIFICANT IMPACT

The Commission has determined not to prepare an environmental impact statement for the proposed license amendments.

Based upon the foregoing environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see the application for amendments dated April 15, 1986, as supplemented July 25 and September 22, 1986 which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

Dated at Bethesda, Maryland, this 23th day of October 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

U. S. NUCLEAR REGULATORY COMMISSION

GEORGIA POWER COMPANY, ET AL.

DOCKETS NOS. 50-321 AND 50-366

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (Commission) has issued Amendments Nos. 123 and 66 to Facility Operating Licenses Nos. DPR-57 and NPF-5, issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia (the licensee), which revised the Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (the facility) located in Appling County, Georgia. The amendments are effective as of the date of issuance and shall be implemented within 60 days.

The amendments a) permit use of Banked Position Withdrawal Sequences for the first 50 percent of control rod withdrawal; b) remove the linear mass restriction of 15.2 grams of Uranium-235 per centimeter for fuel assemblies stored in the fuel pool; c) eliminate specific mechanical descriptions of fuel assemblies; d) provide Maximum Average Planar Linear Heat Generation limit curves for several new fuel assemblies; and, e) make several editorial changes.

The application for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the

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Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendments and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on June 30, 1986 (51 FR 23611). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment and Finding of No Significant Impact related to the action and has concluded that an environmental impact statement is not warranted because there will be no environmental impact attributable to the action significantly beyond that which has been predicted and described in the Commission's Final Environmental Statement for the facility.

For further details with respect to the action see (1) the application for amendments dated April 15, as supplemented July 25, 1986, (2) Amendments Nos. 123 and 66 to Facility Operating Licenses Nos. DPR-57 and NPF-5, (3) the Commission's related Safety Evaluation, and (4) the Environmental Assessment dated October 23, 1986. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of BWR Licensing.

Dated at Bethesda, Maryland this 31st day of October, 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing