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.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.G.2. <u>Rod Sequence Control System (RSCS)</u>	2. <u>Rod Sequence Control System (RSCS)</u>
a. <u>Operability</u> 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.	a. <u>Operability</u> 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. <u>Failed Position Switch</u> 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).	b. <u>Failed Position Switch</u> 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.

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.

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.

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-

3.11. 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 $\geq$ 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 $\geq$ 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% $\leq$ power < 30%, the OLMCPR is given in Figure 3.11.6. For power $\geq$ 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

$\tau_A = 0.90$ sec (Specifications 3.3.C.2.a. scram time limit to 20% insertion from fully withdrawn)

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

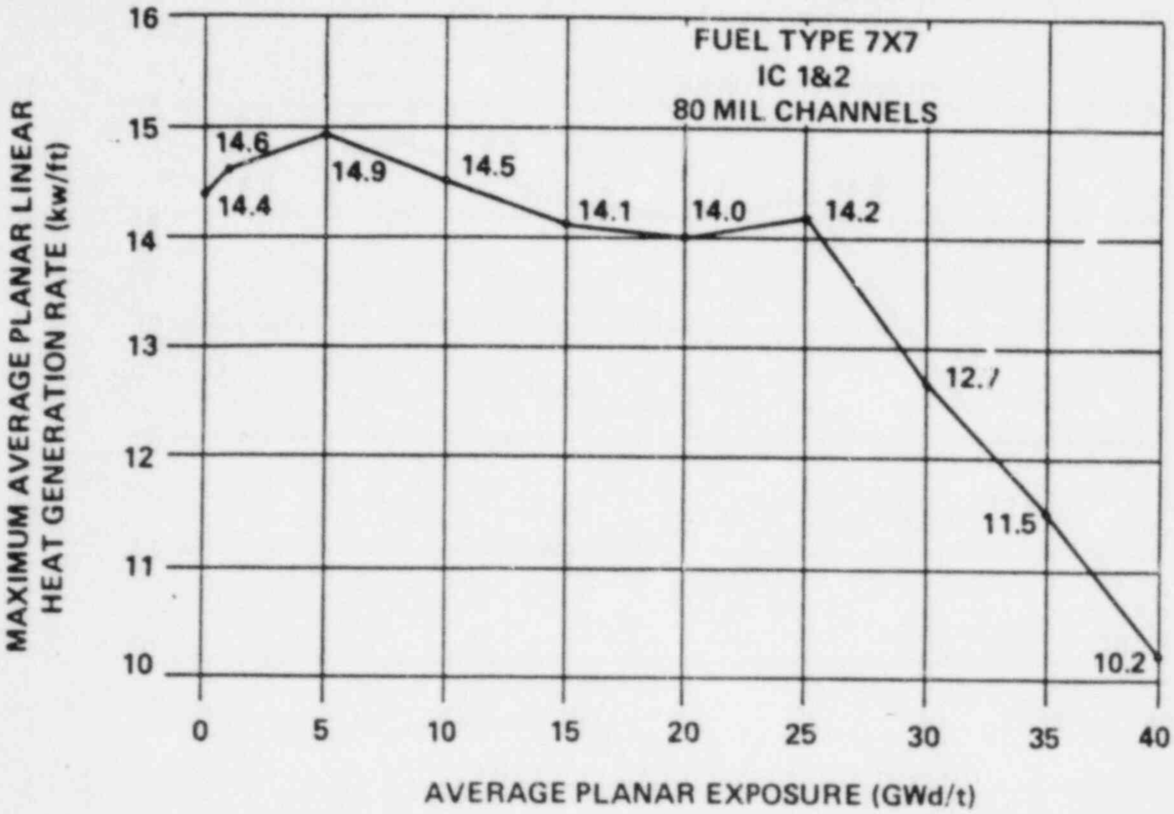
3.11.C. Minimum Critical Power Ratio (MCPR) (Continued)

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])



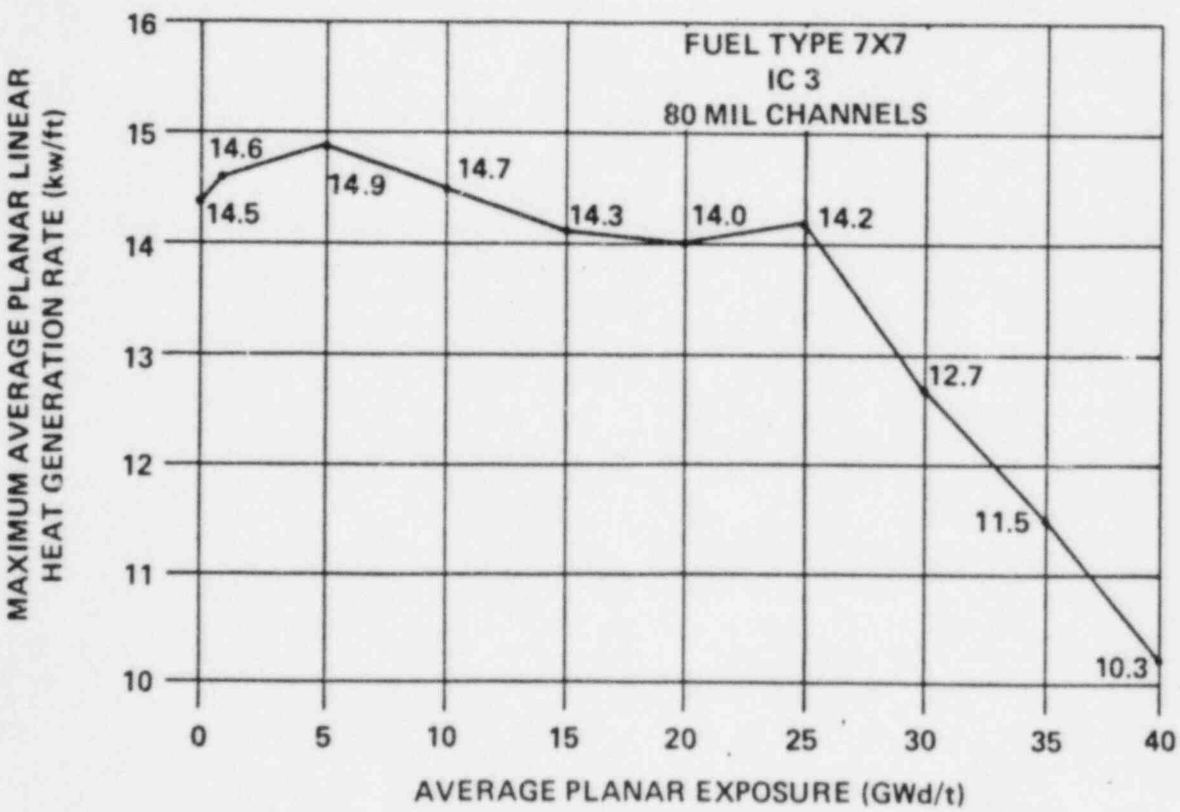
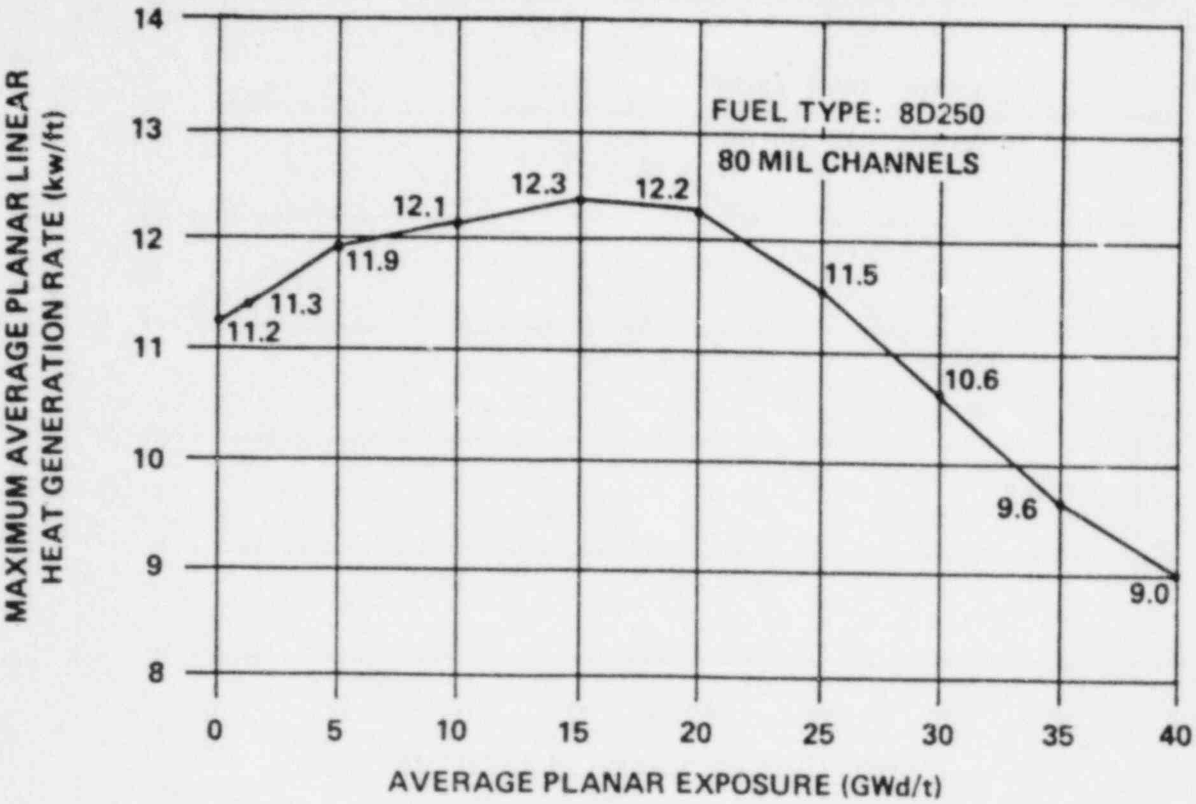


FIGURE 3.11-1 (SHEET 1)



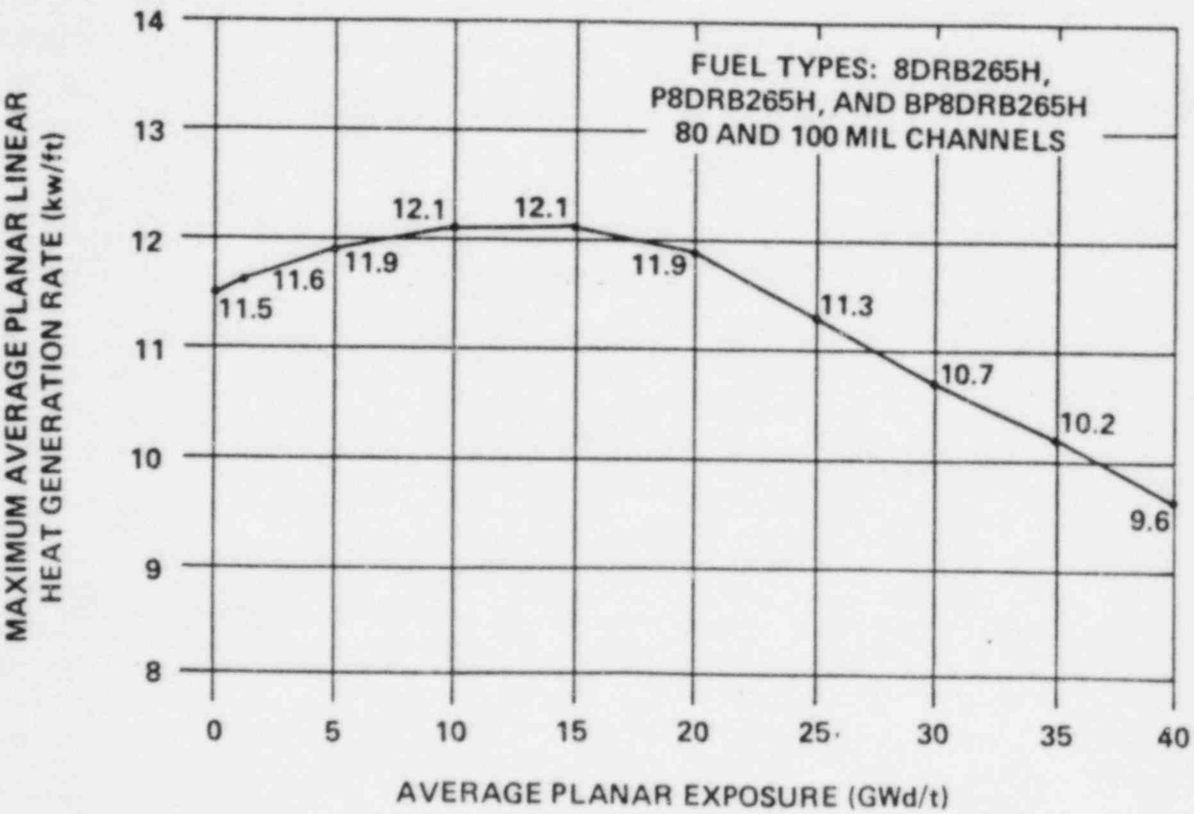
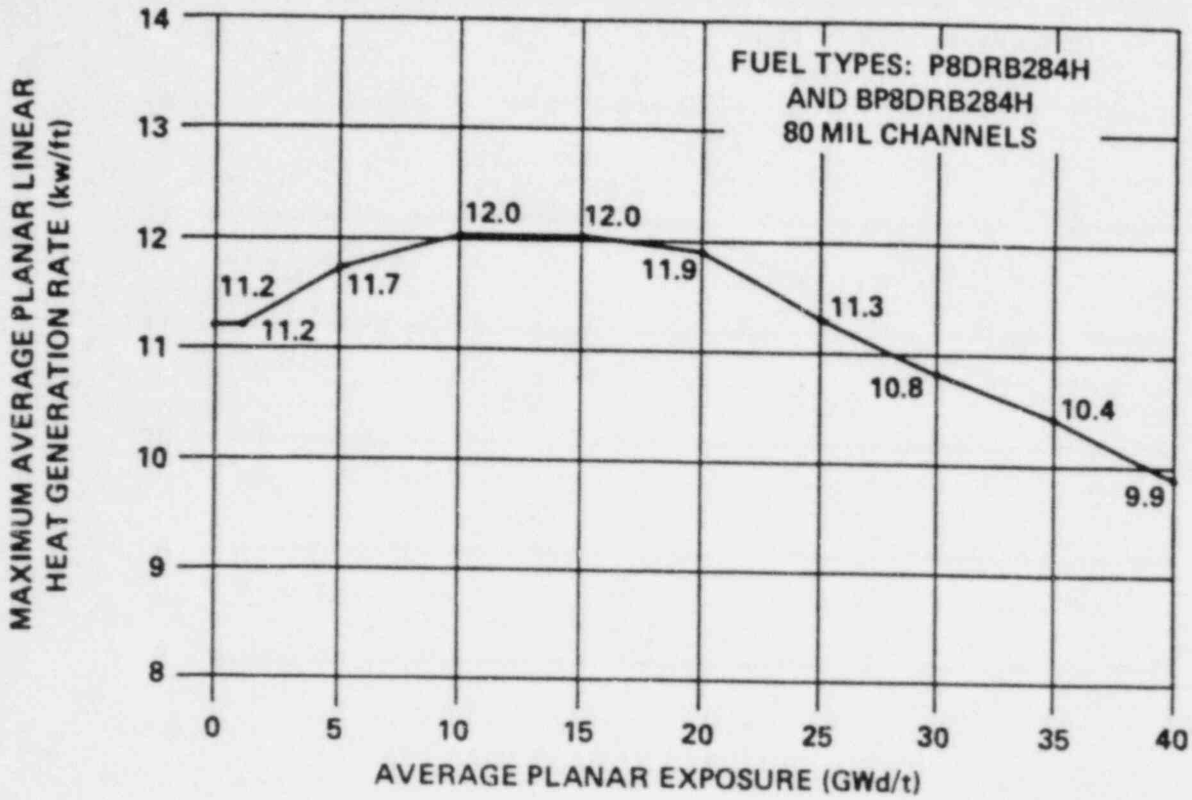


FIGURE 3.11-1 (SHEET 2)



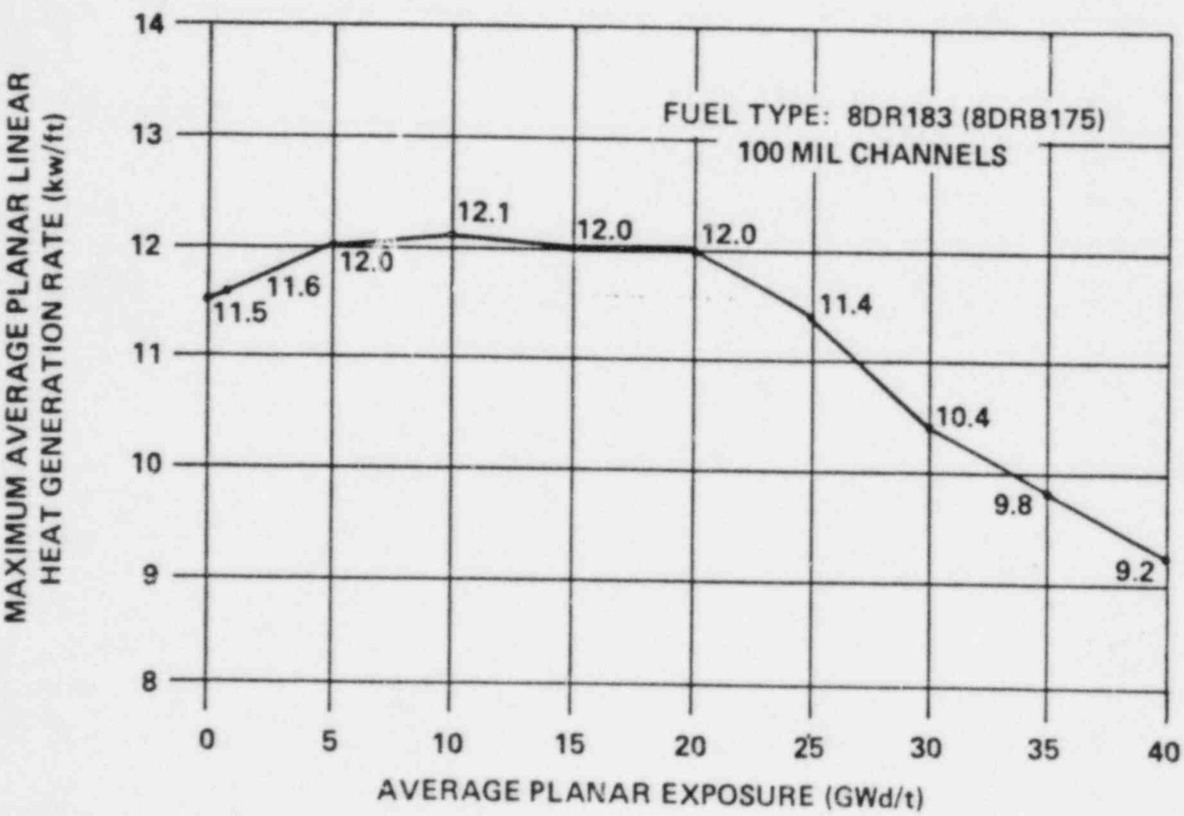
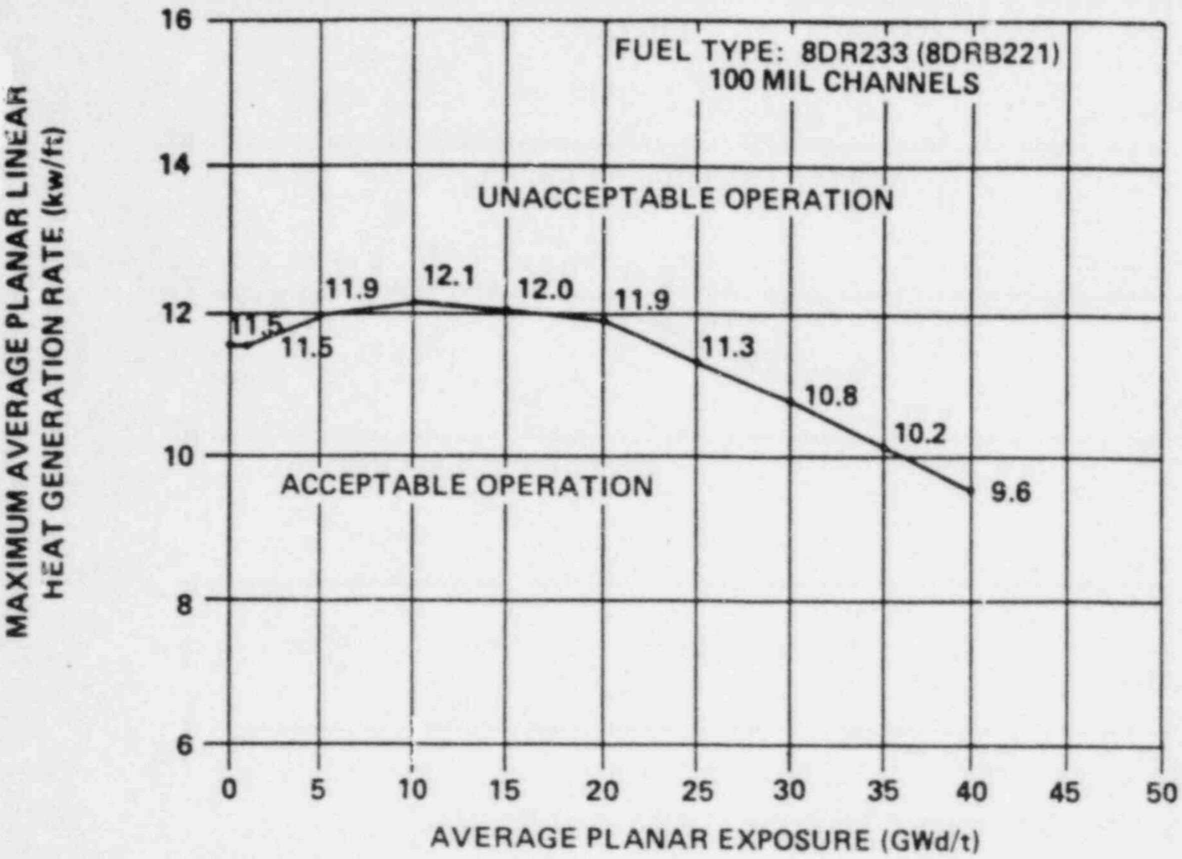


FIGURE 3.11-1 (SHEET 3)



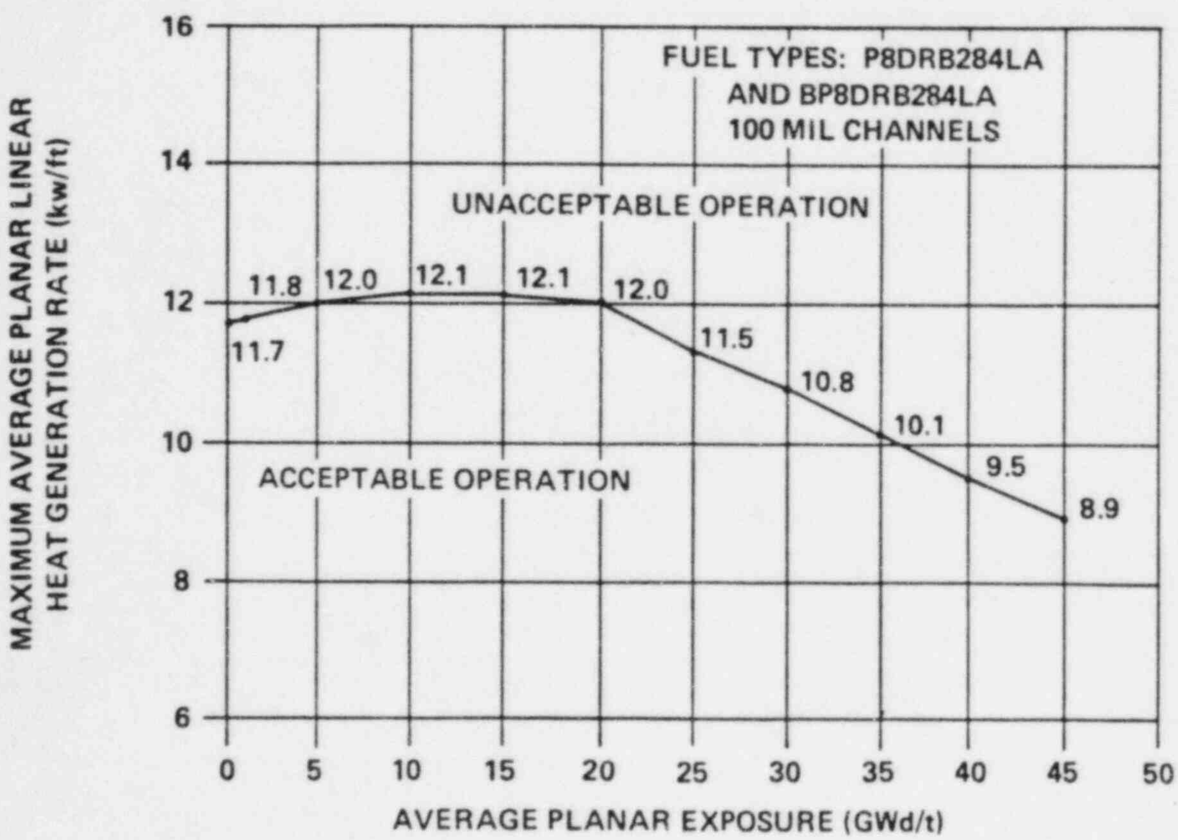
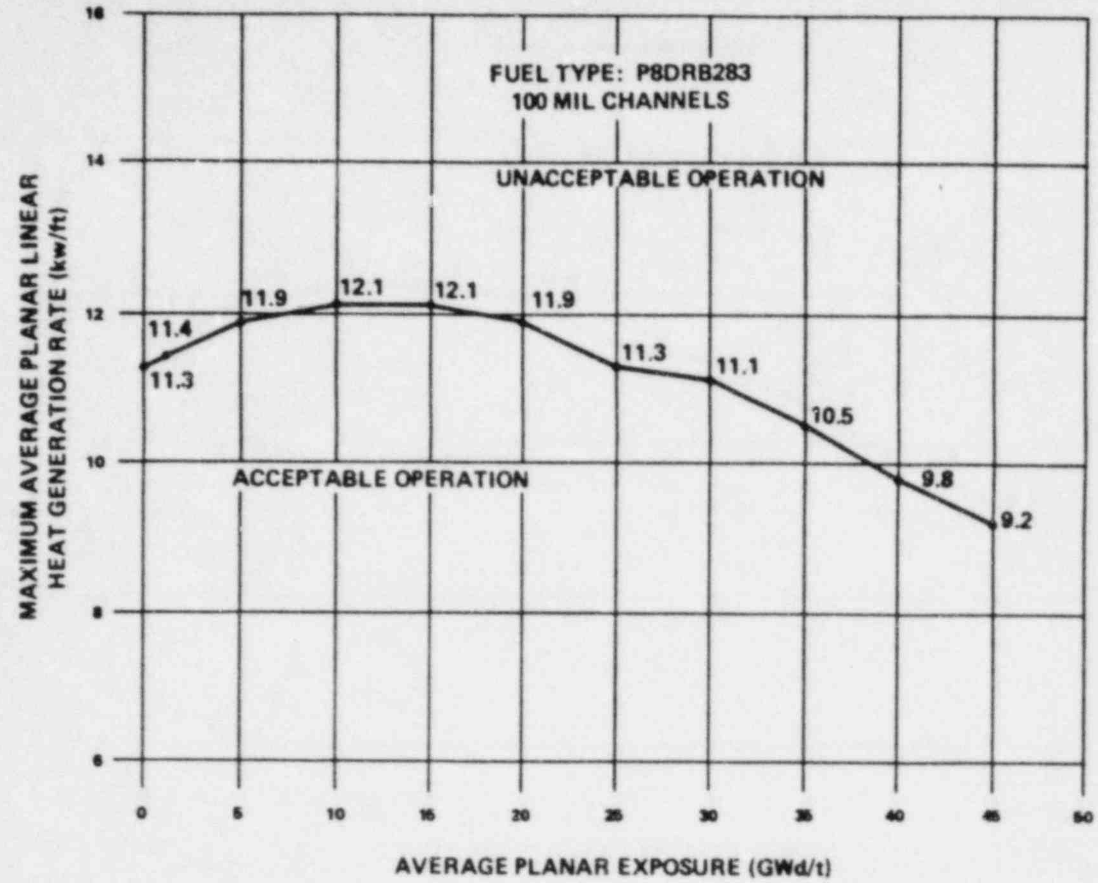


FIGURE 3.11-1 (SHEET 4)



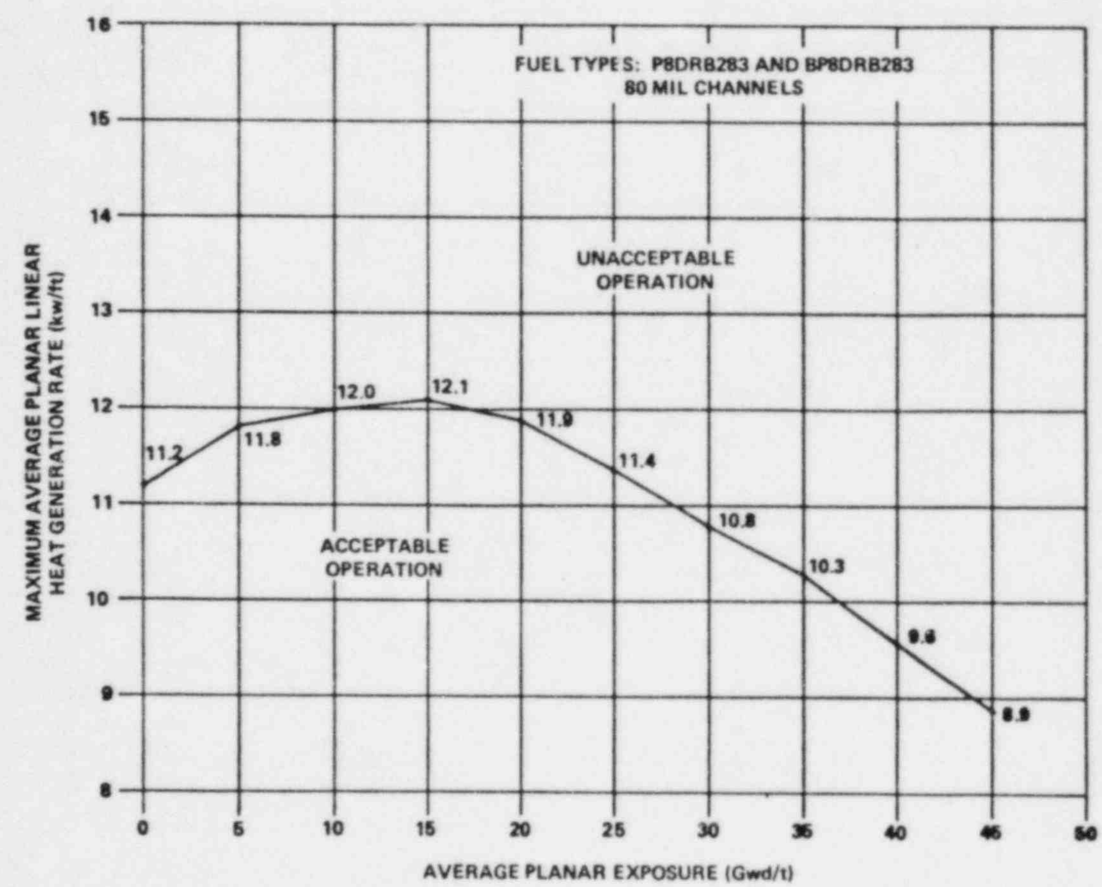


FIGURE 3.11-1 (SHEET 5)

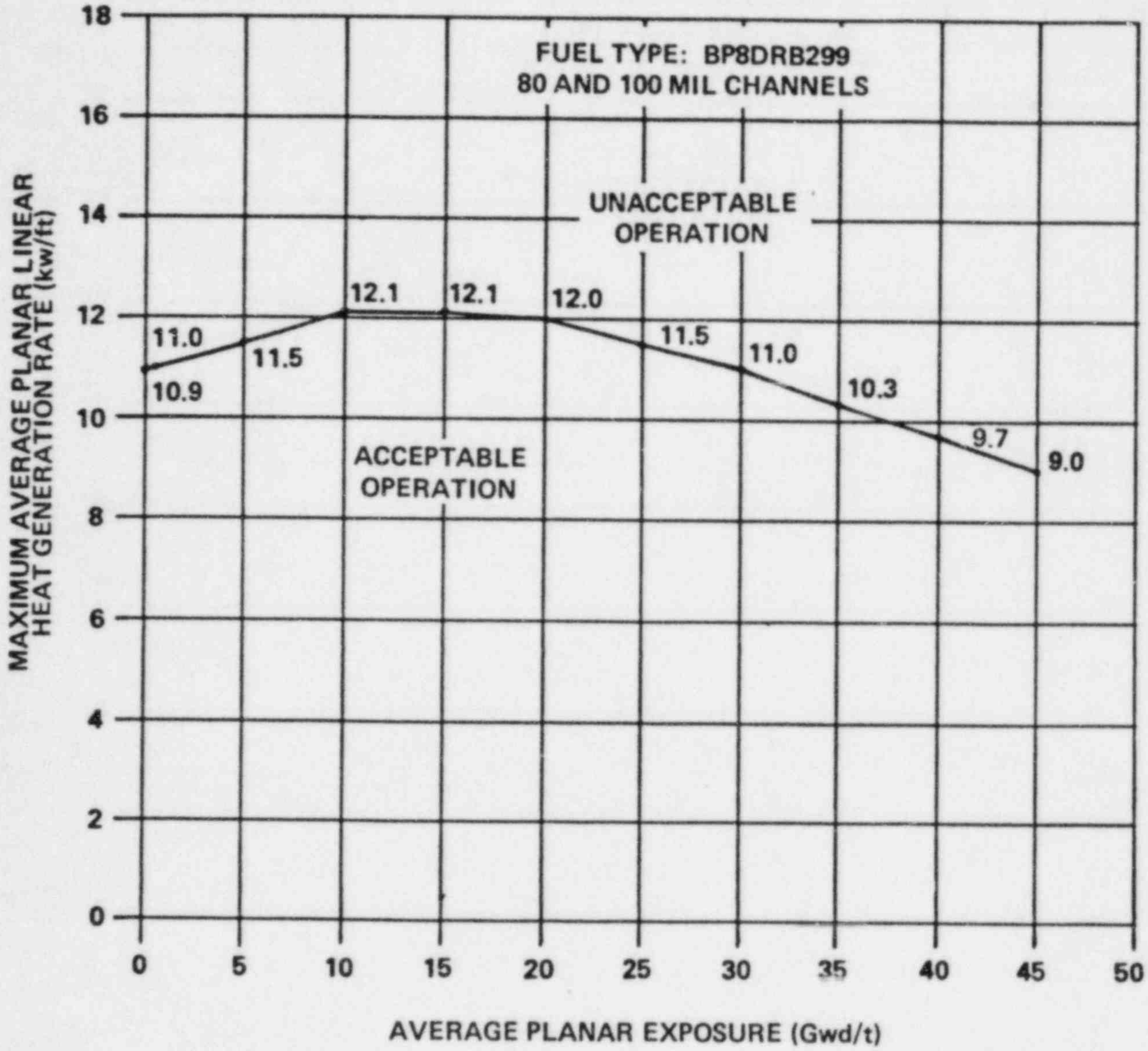


FIGURE 3.11-1 (SHEET 6)

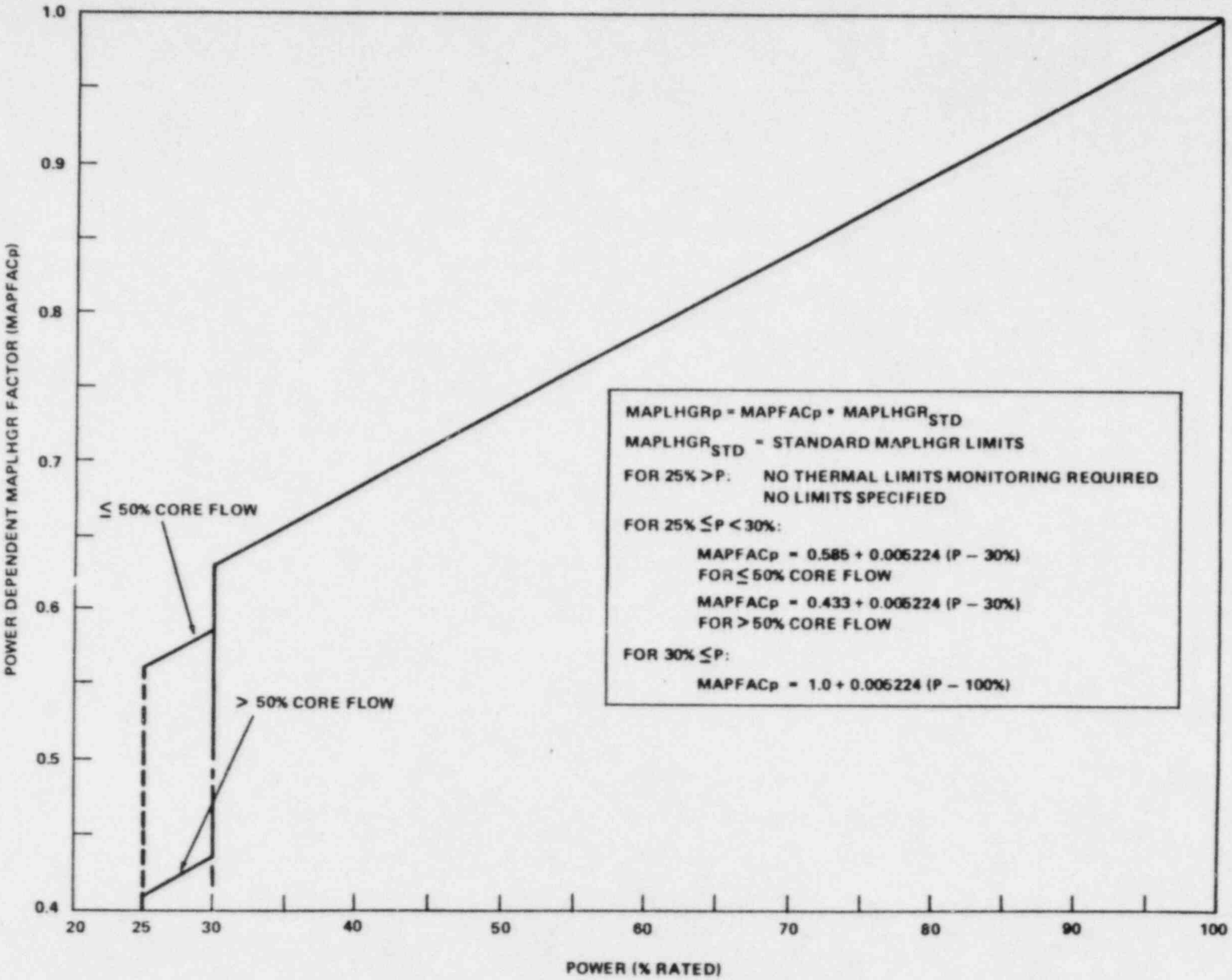


FIGURE 3.11-1 (SHEET 7) MAPFACp

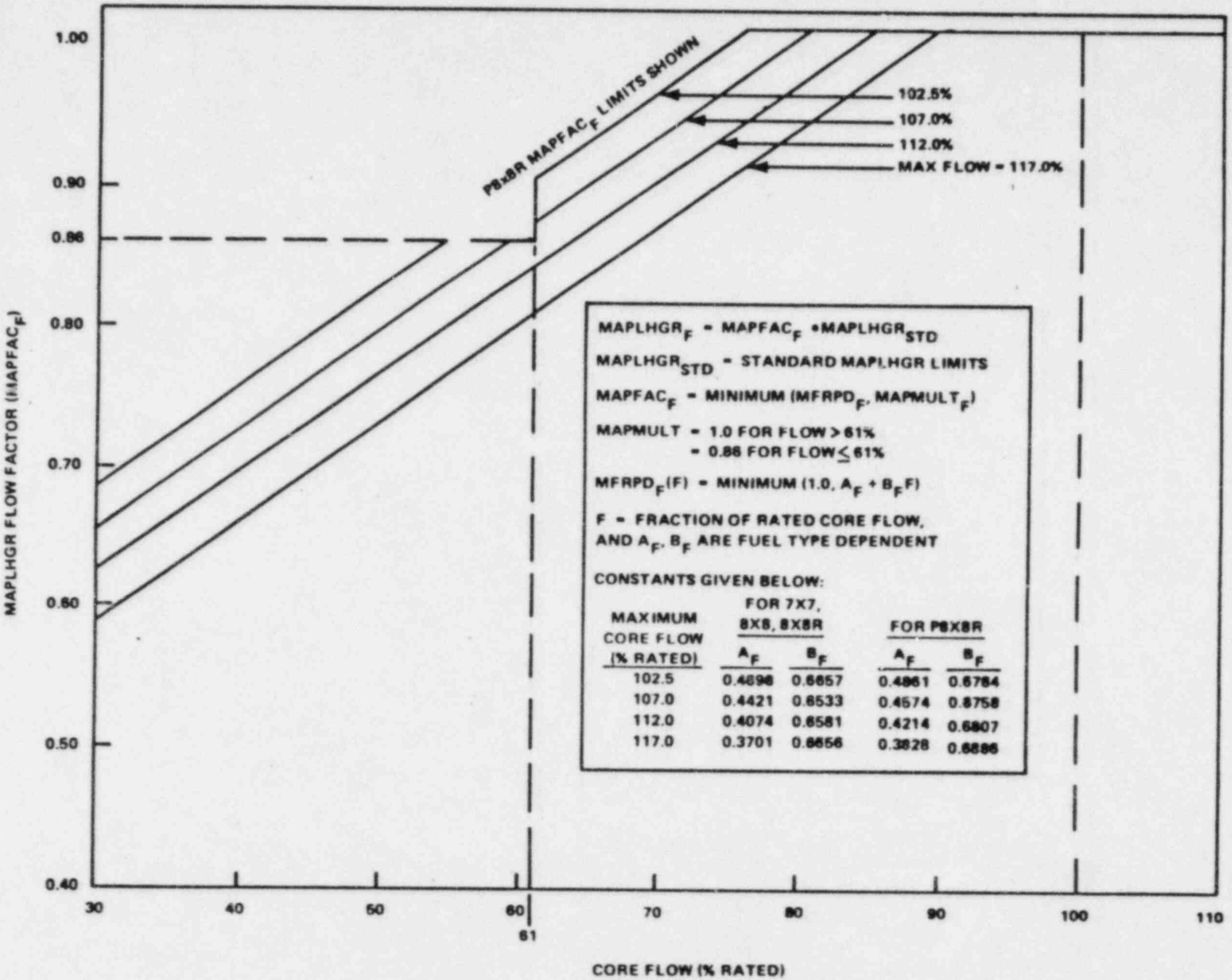


FIGURE 3.11-1 (SHEET 8) MAPFAC$_F$

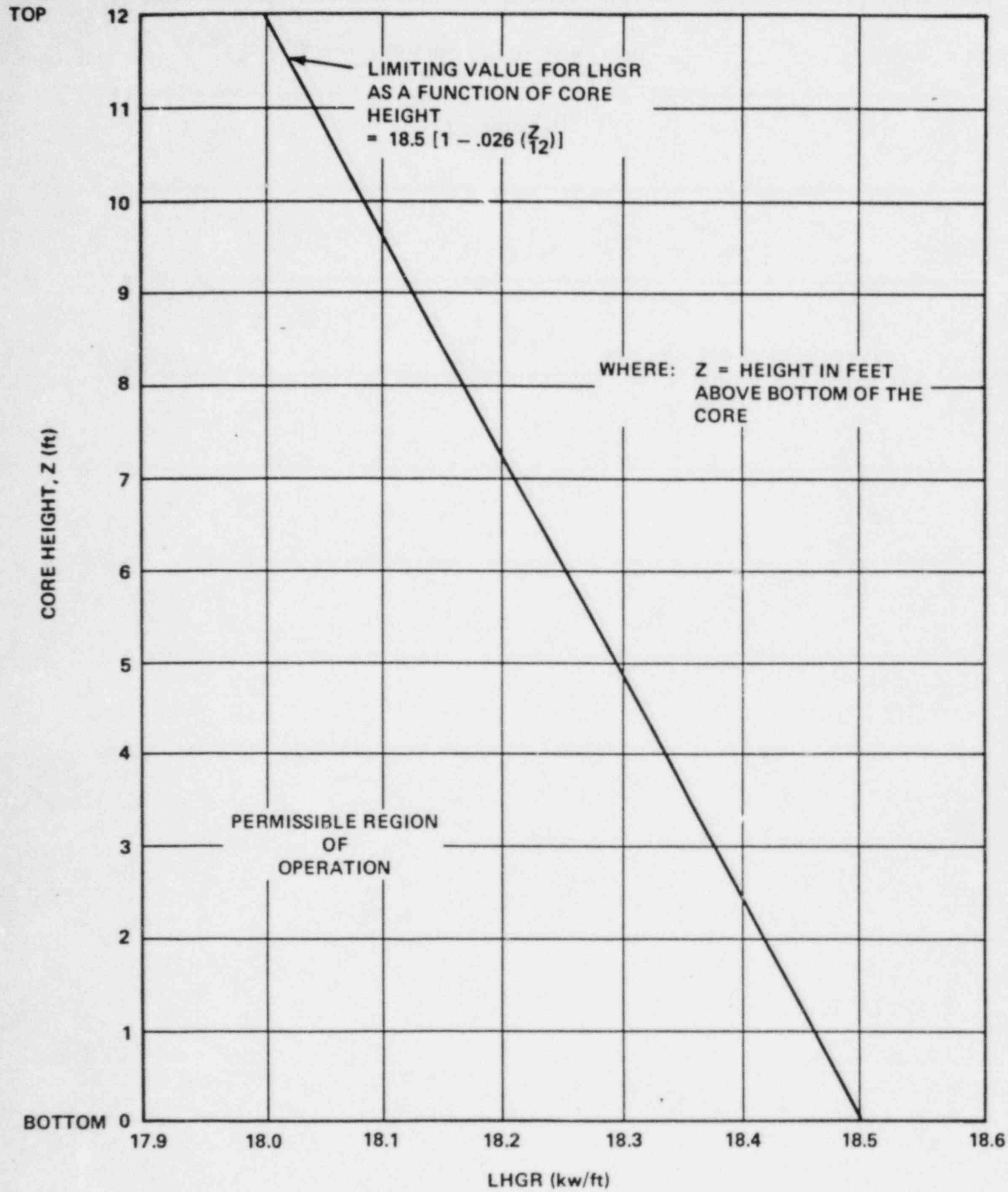


FIGURE 3.11-2
LIMITING VALUE FOR LHGR
FUEL TYPE 7X7

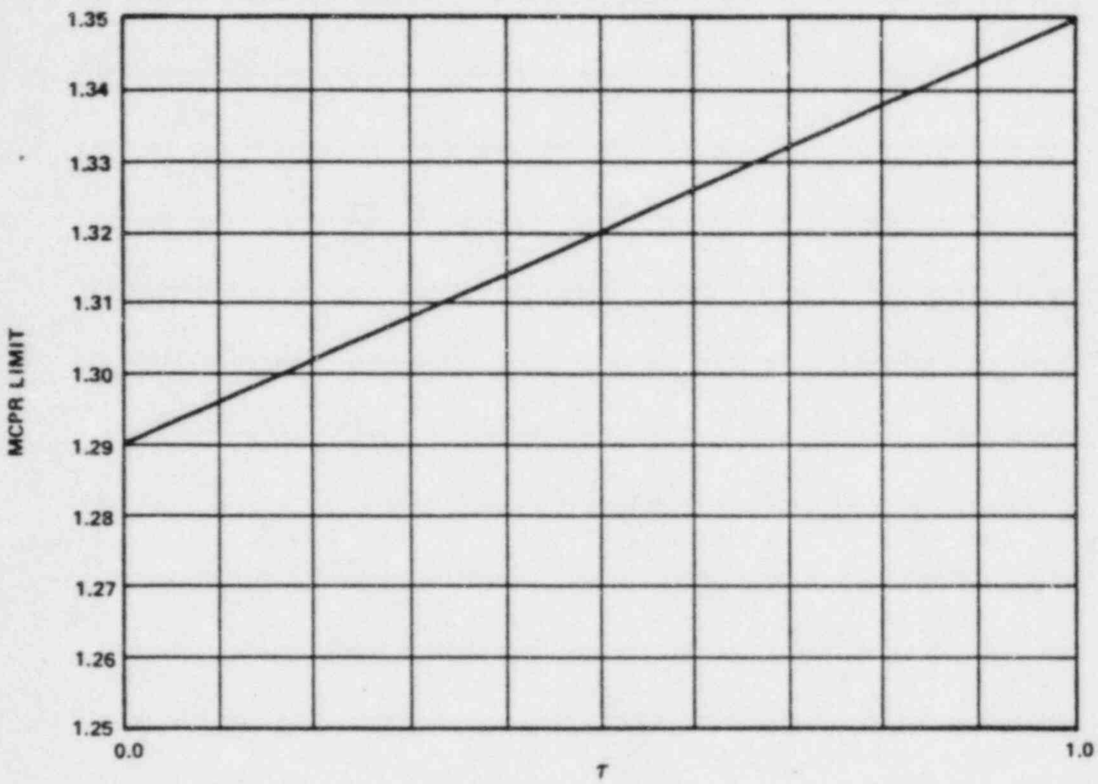


FIGURE 3.11.4
MCPR LIMIT FOR ALL 8X8 FUEL TYPES
FOR RATED POWER AND RATED FLOW

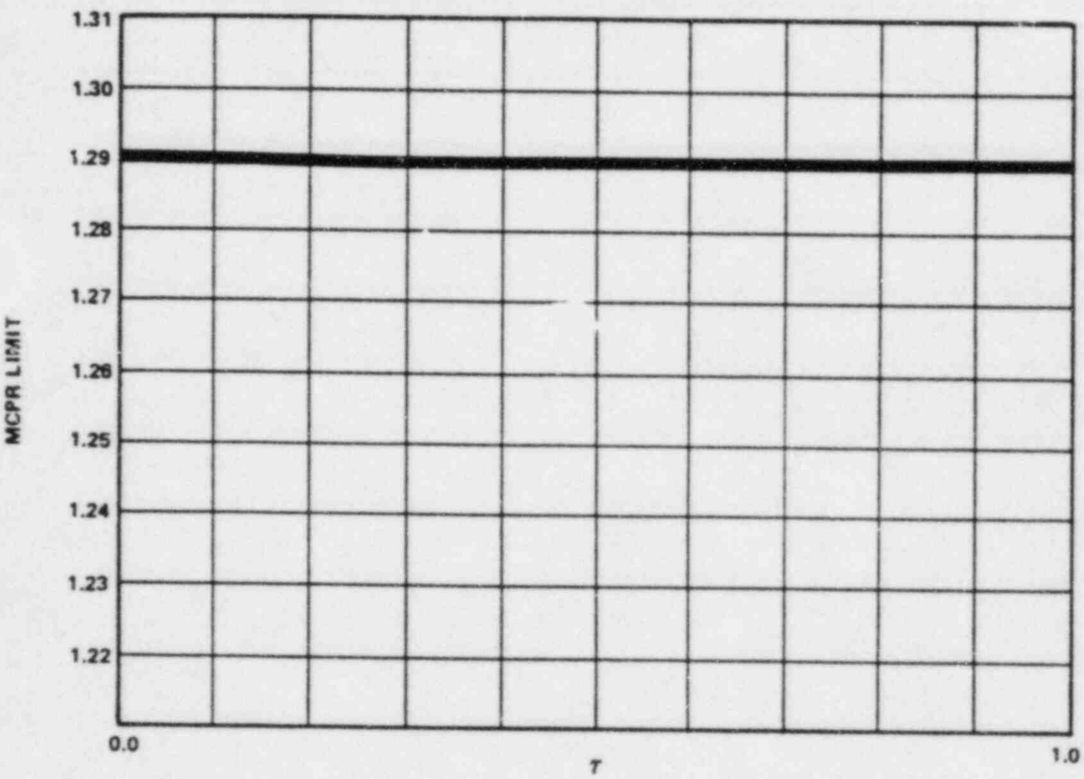


FIGURE 3.11-5
MCPR LIMIT FOR 7X7 FUEL
FOR RATED POWER AND RATED FLOW

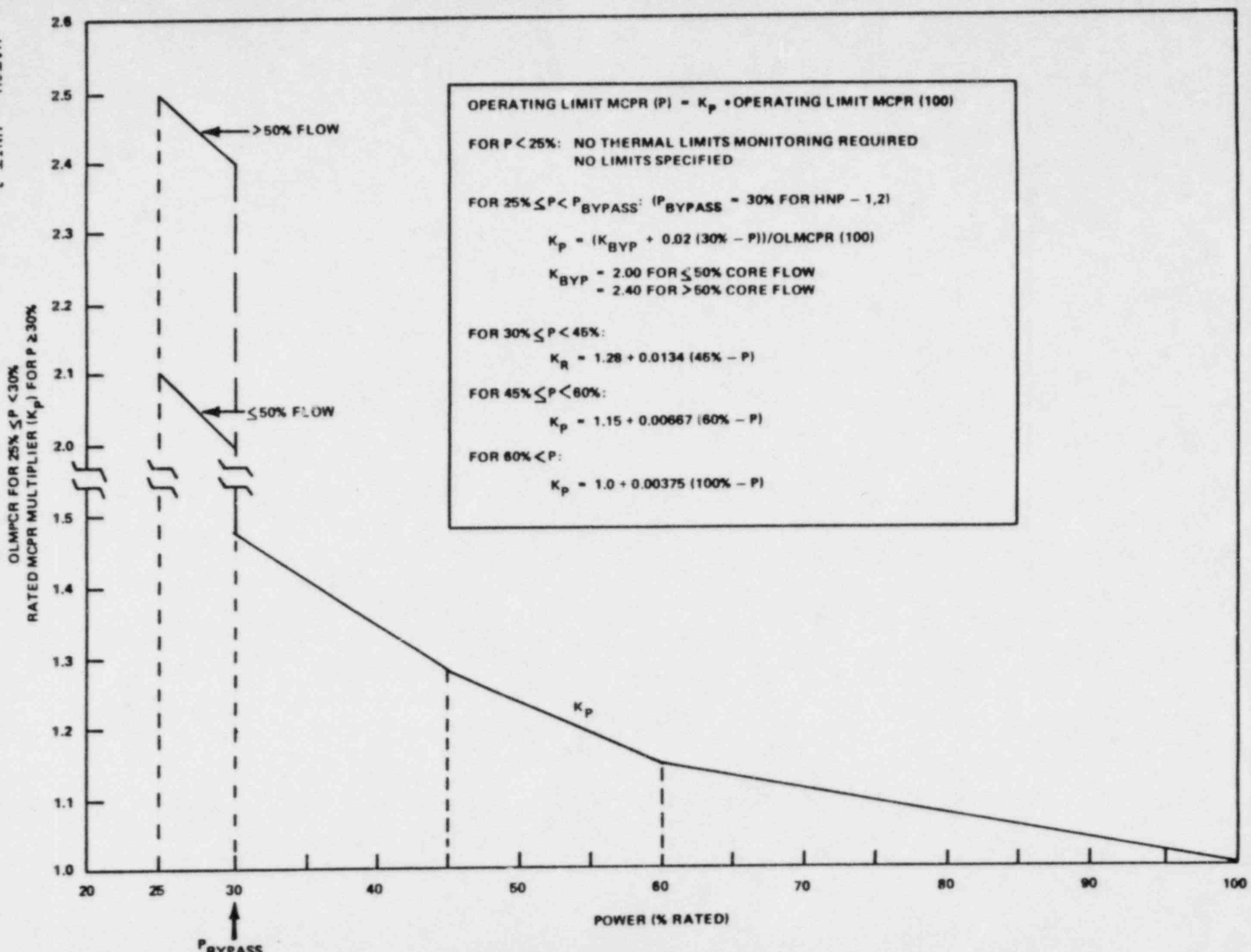


FIGURE 3.11-6 KP

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. 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

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.

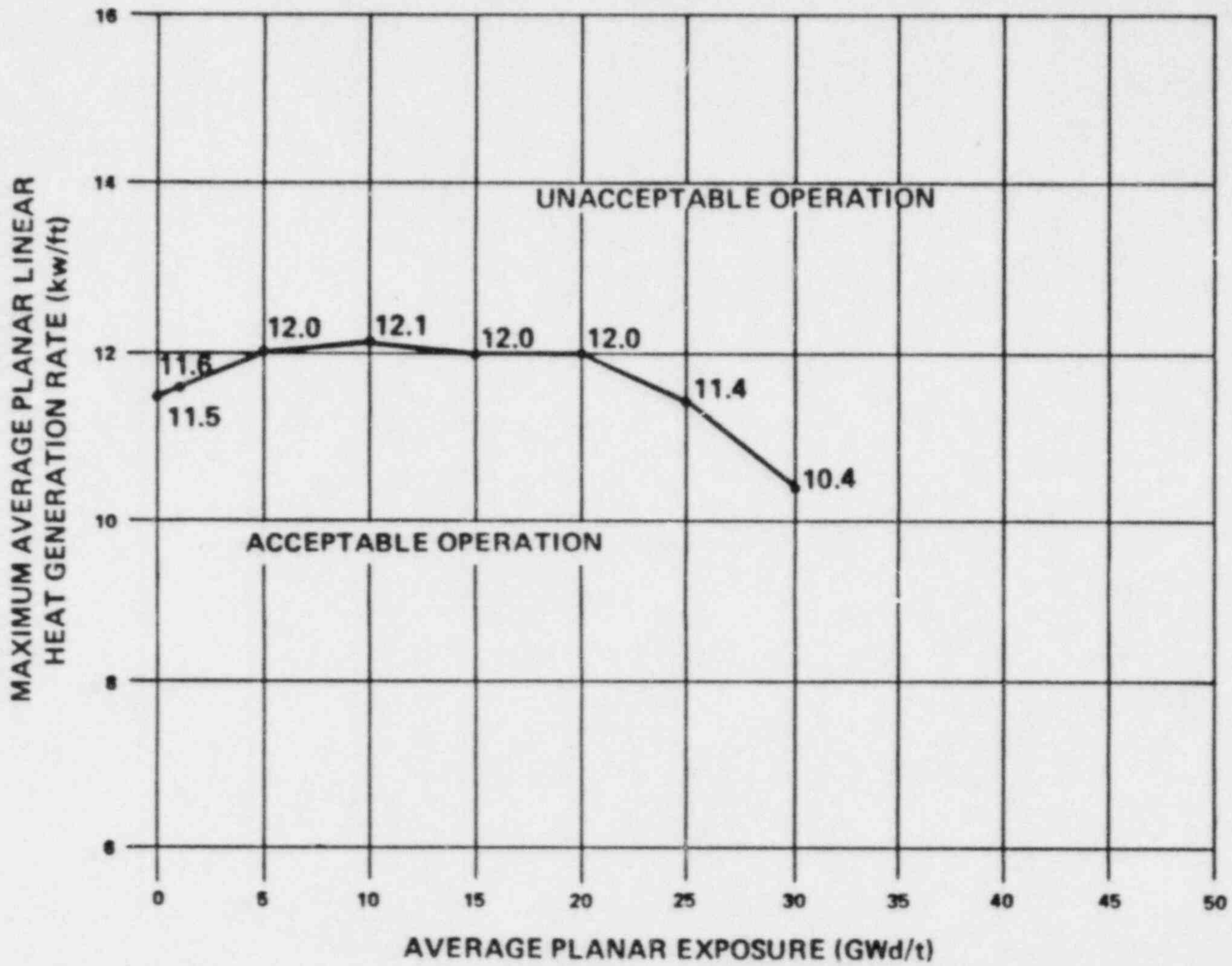
SURVEILLANCE REQUIREMENTS

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:

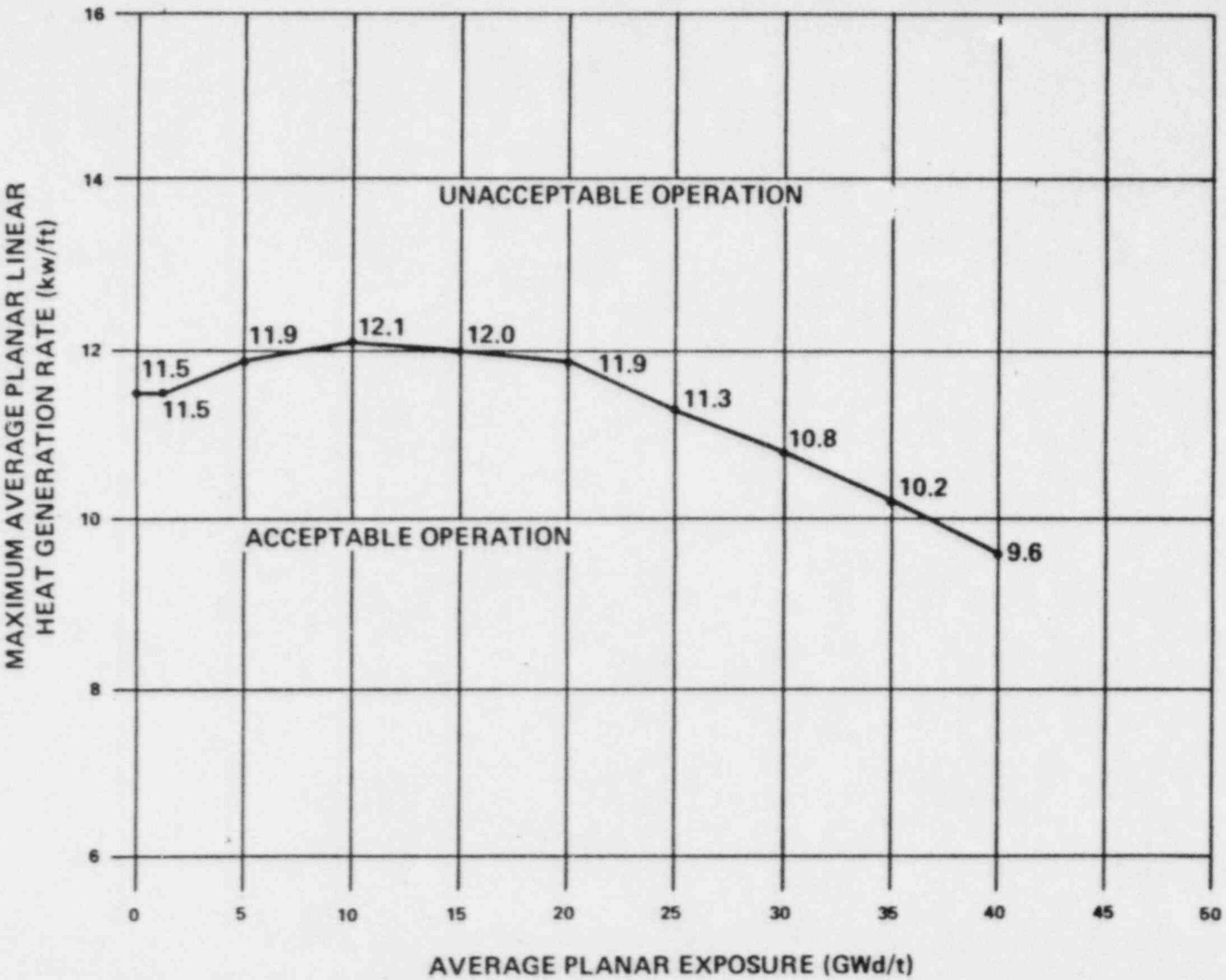
a. At least once per 24 hours,

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

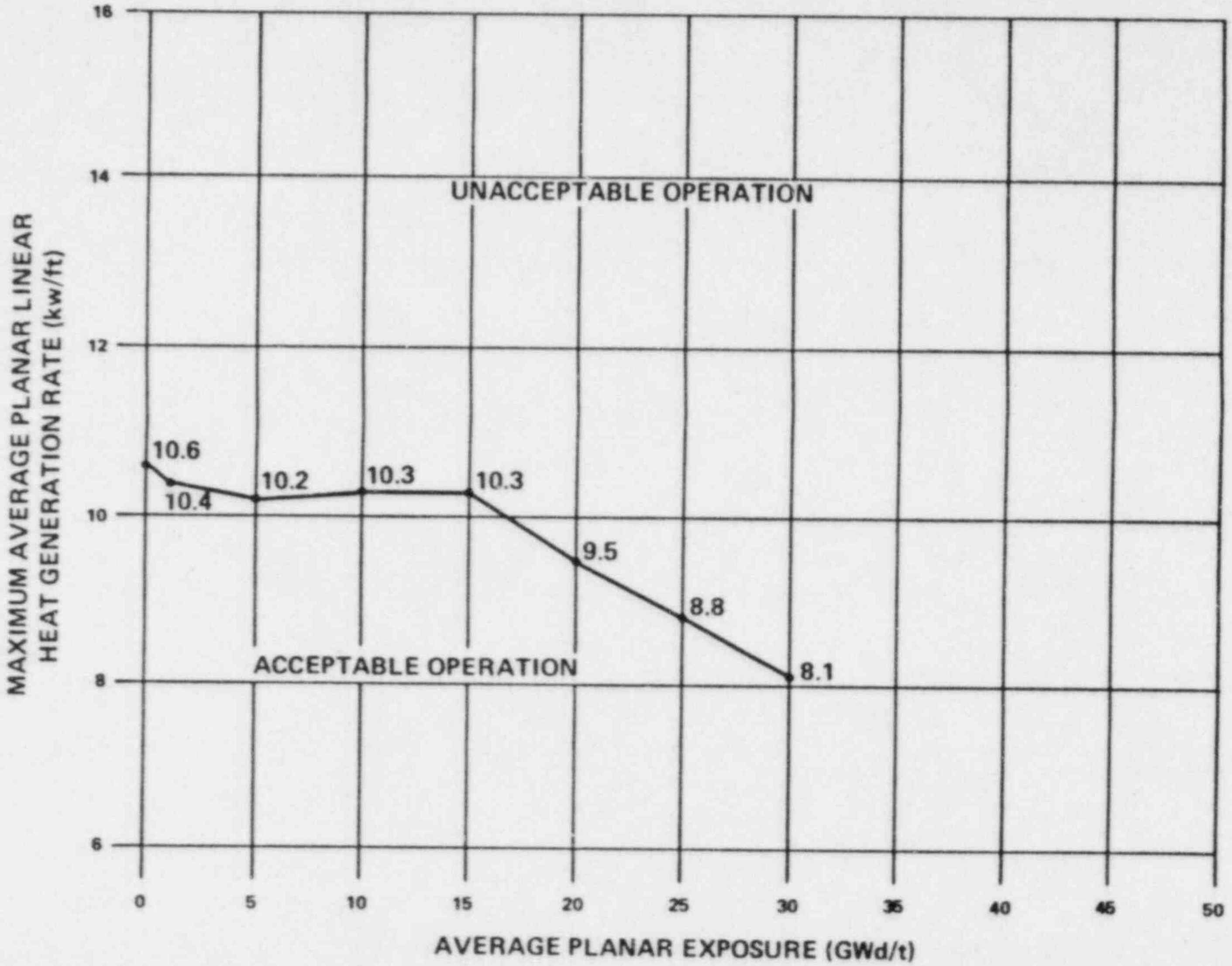
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.



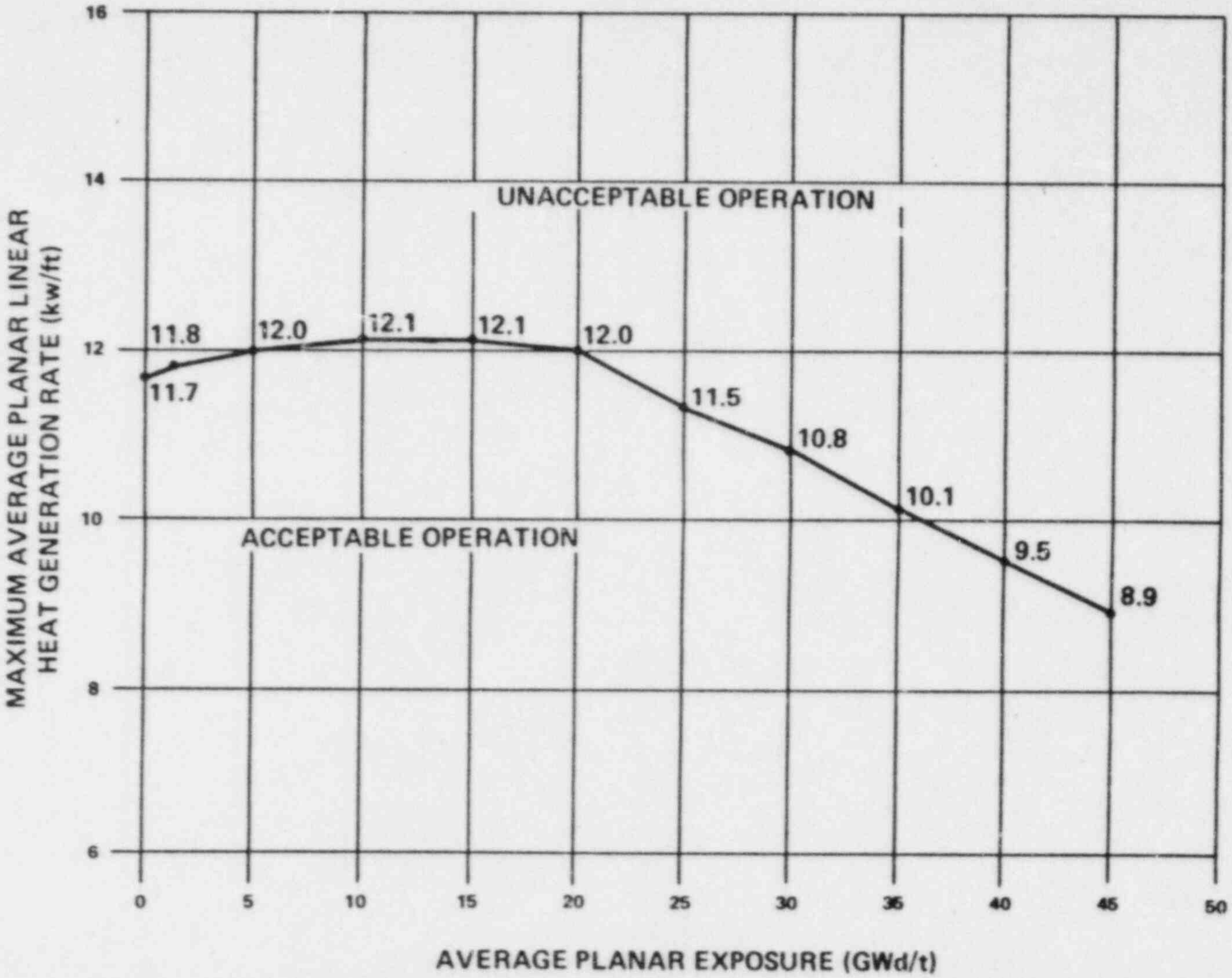
FUEL TYPE 8DIB175 (8DRL183) 100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-1



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

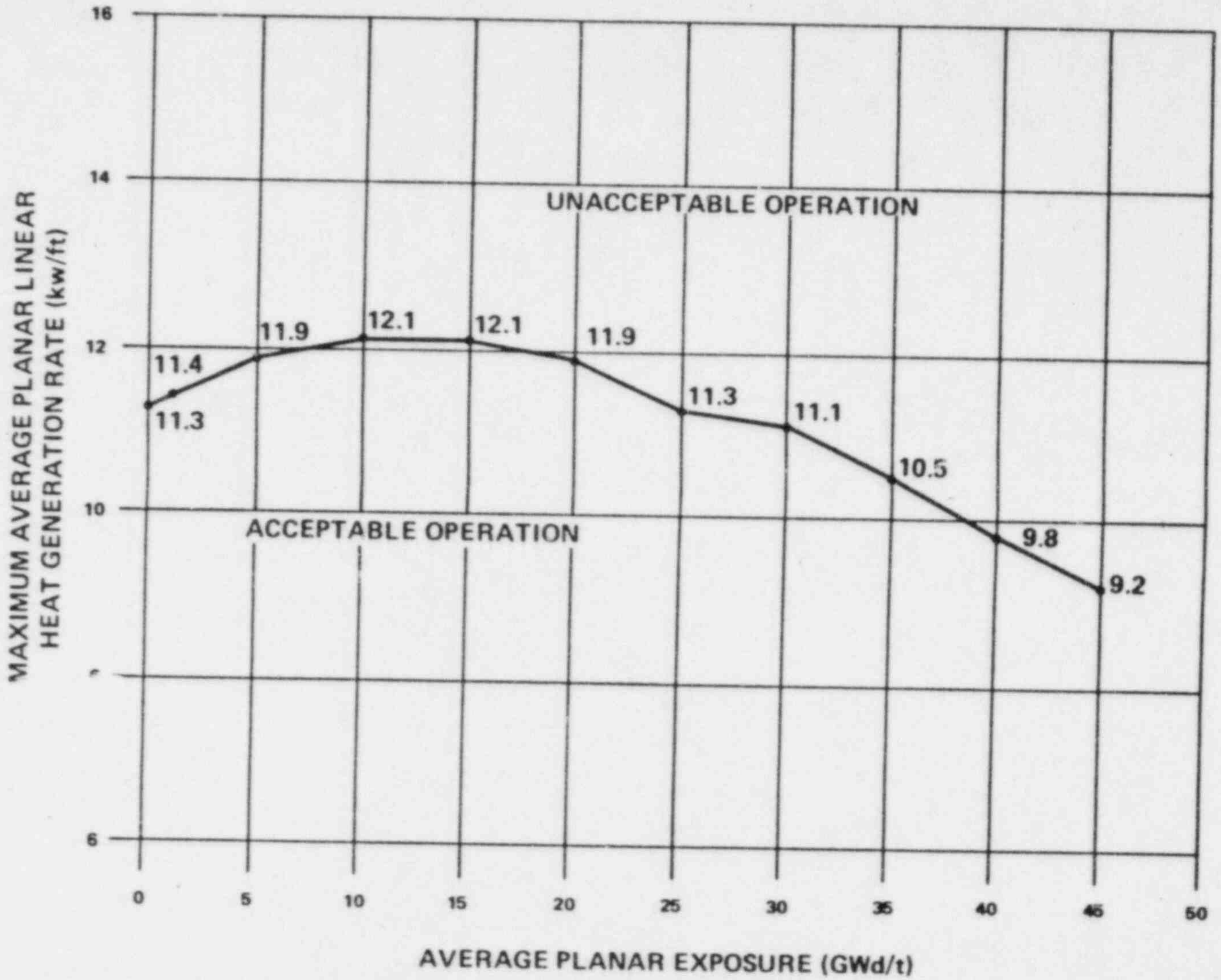


FUEL TYPE IE 711-00GD-100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-3

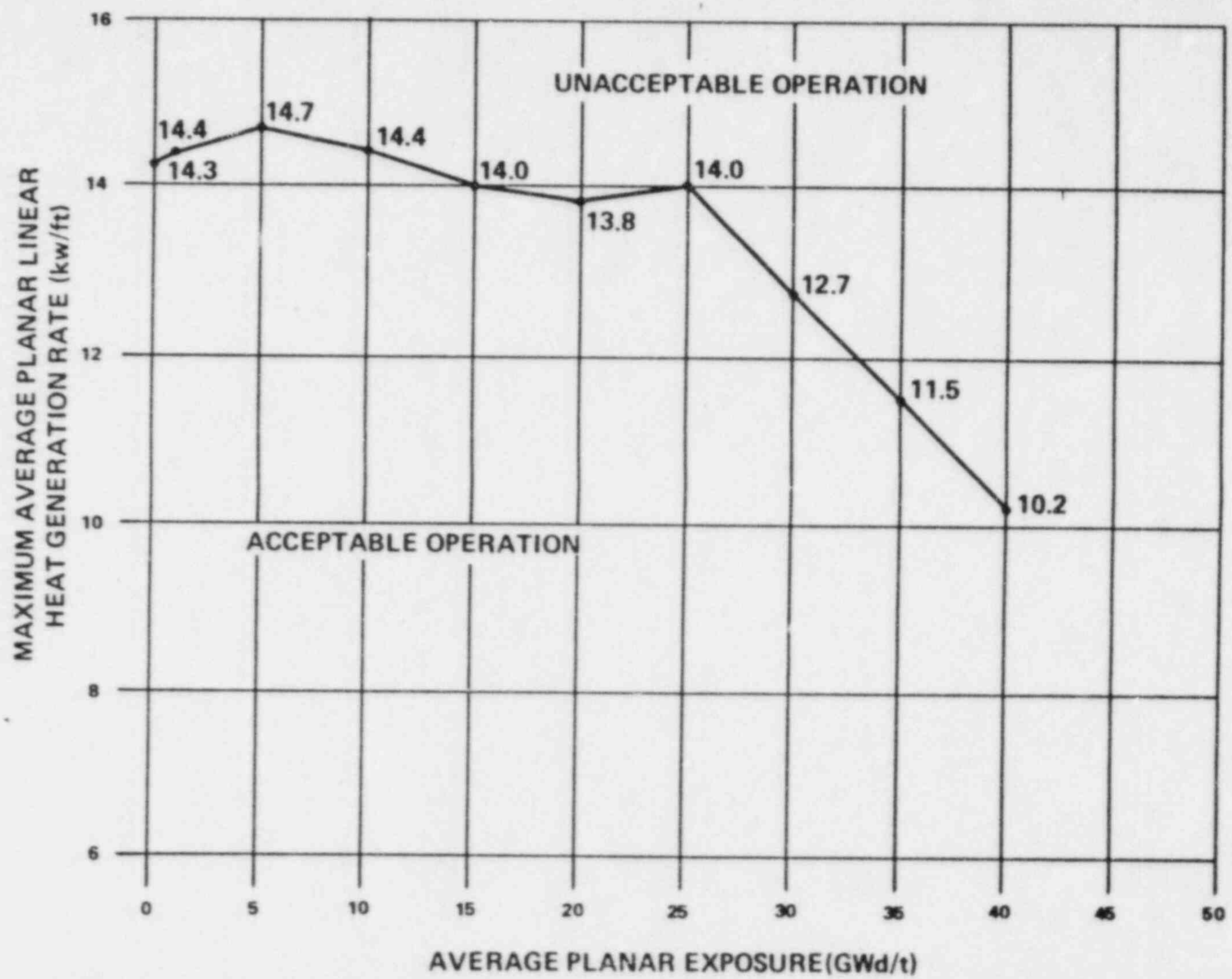


FUEL TYPES P8DRB284LA AND BP8DRB284LA 100 MIL CHANNELS MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR EXPOSURE

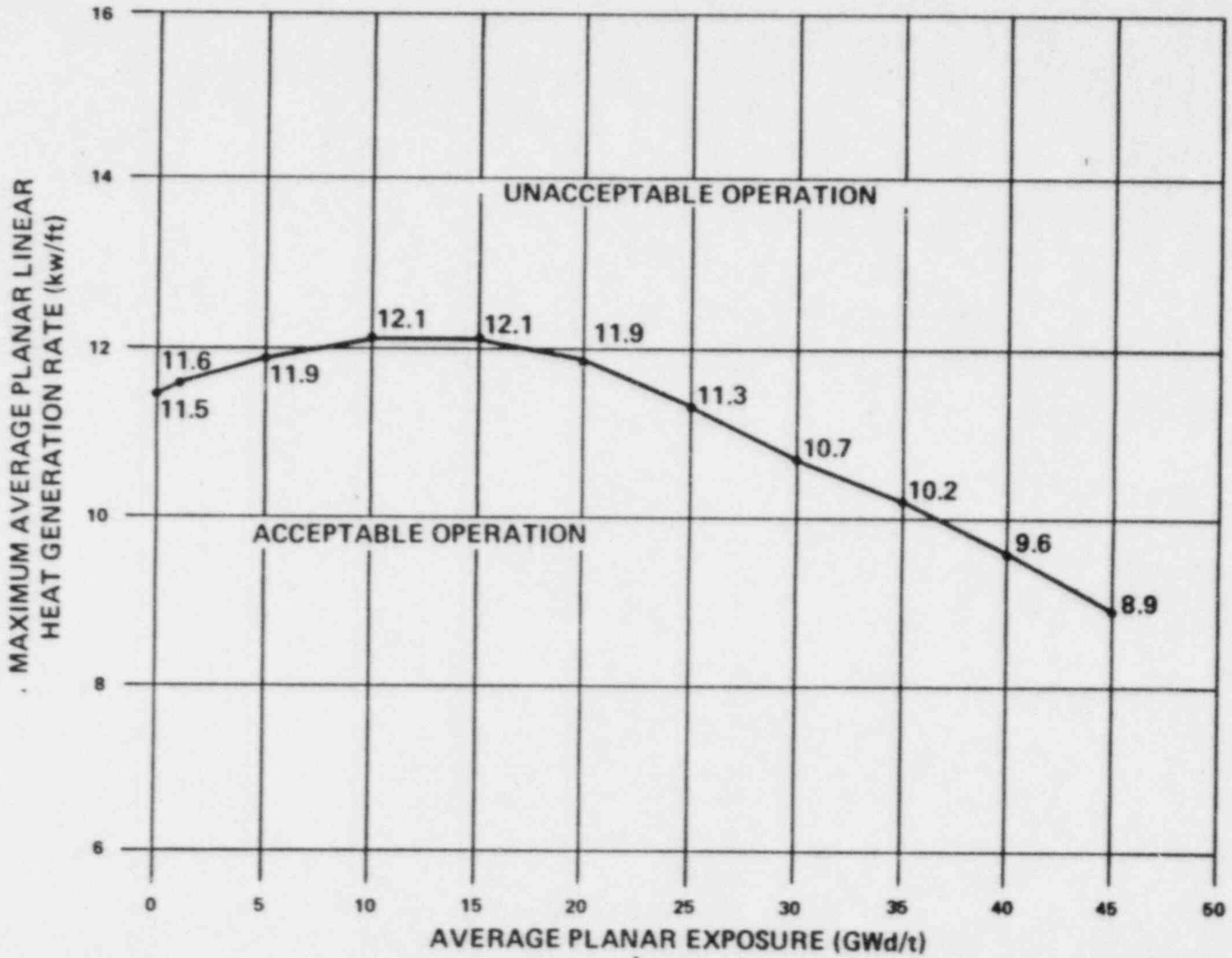
FIGURE 3.2.1-4



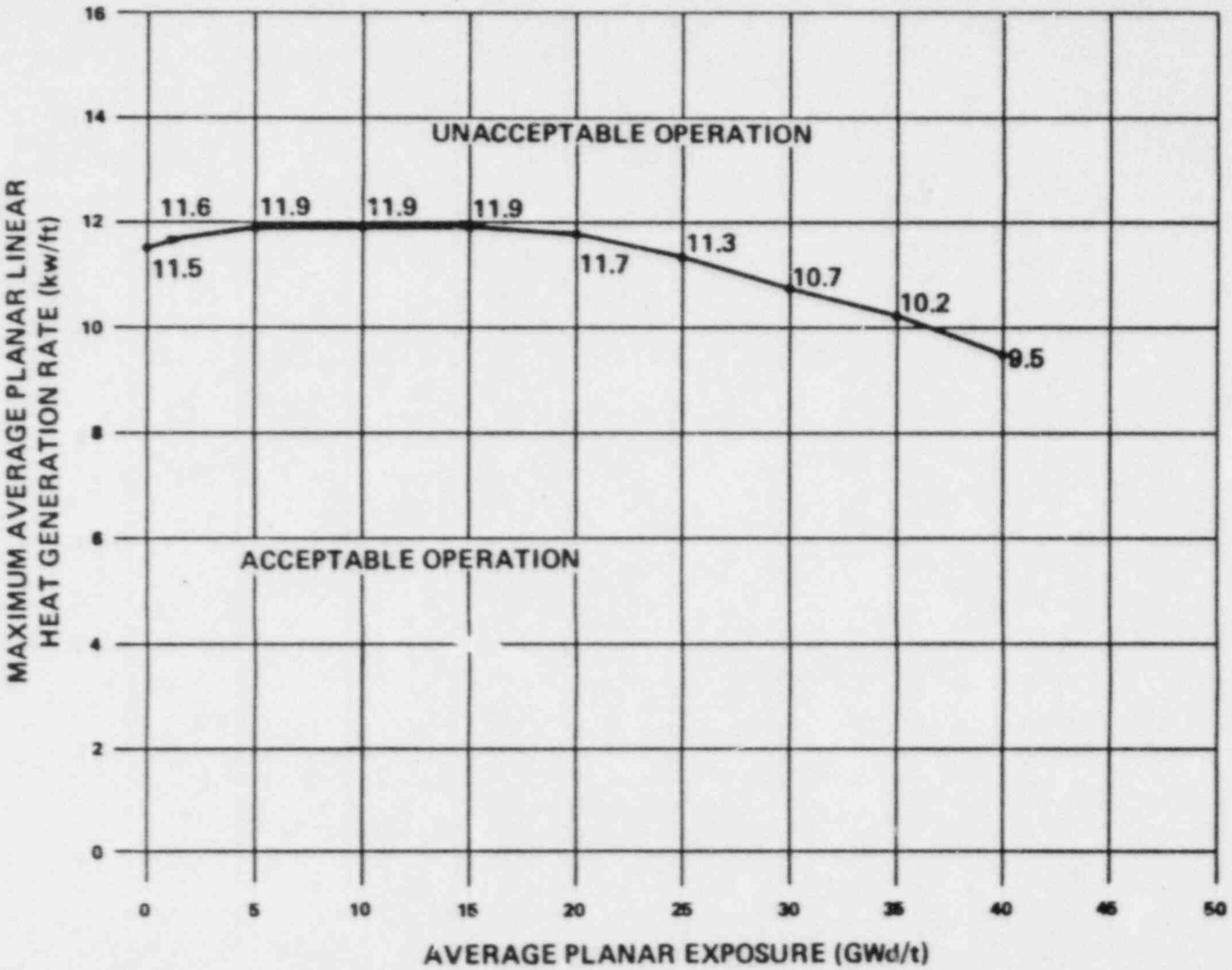
FUEL TYPES P8DRB283 AND BP8DRB283 100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-5



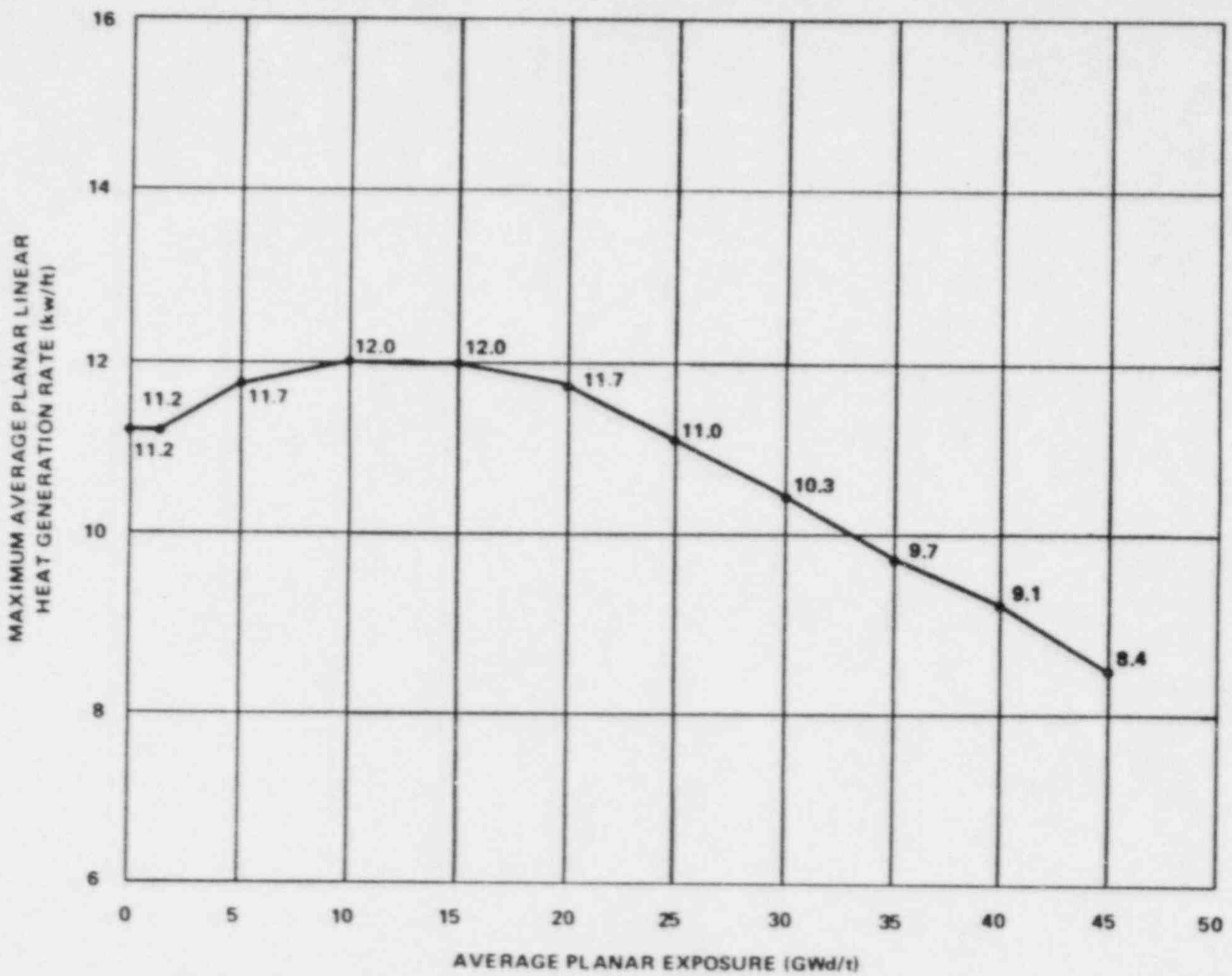
FUEL TYPES HATCH-1 I.C. 1,2,3 (7X7) 80 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MPALHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-6



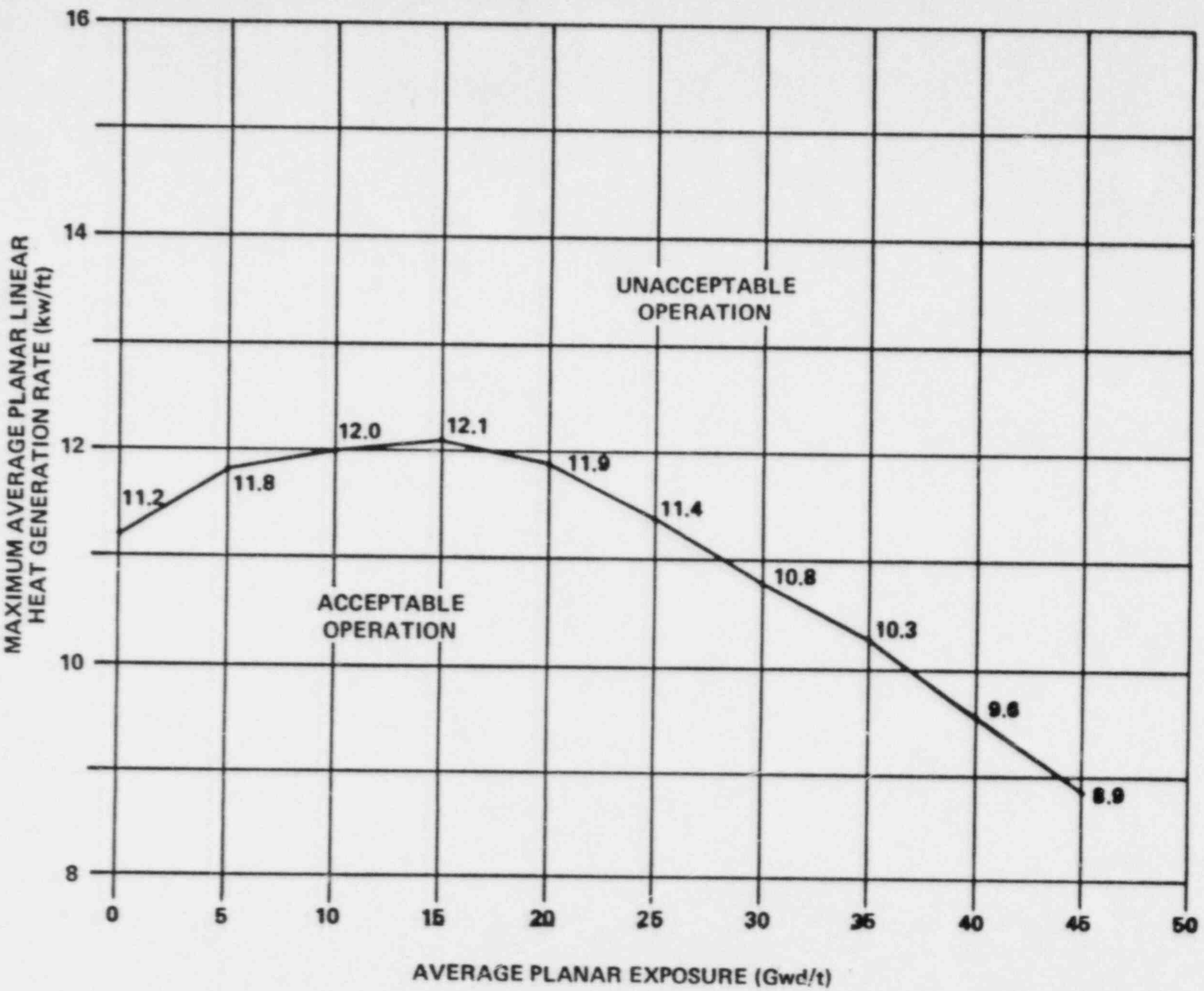
FUEL TYPES P8DRB265H AND BP8DRB265H 80 AND 100 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-7



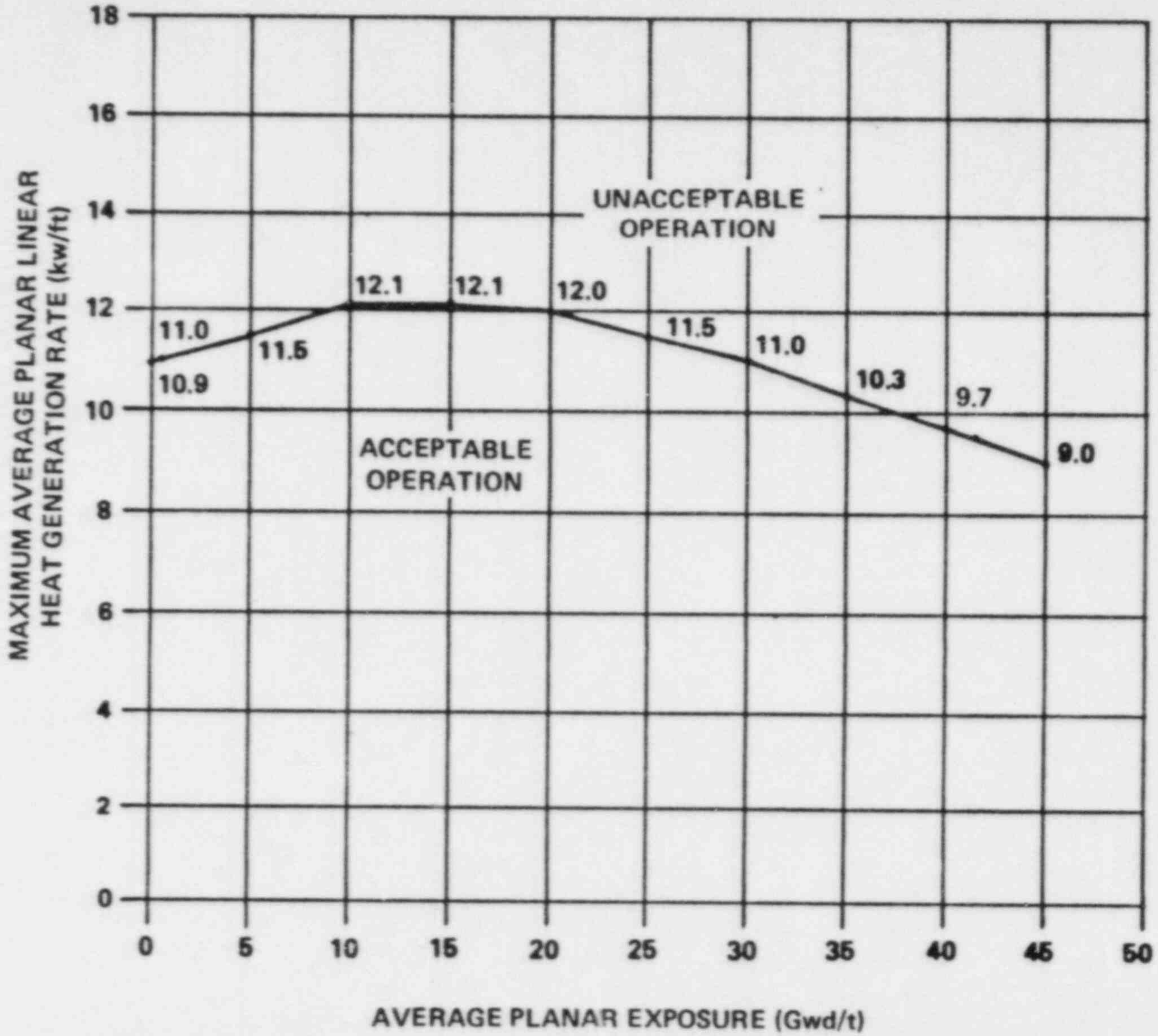
FUEL TYPE 8DRB265H 80 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-8



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

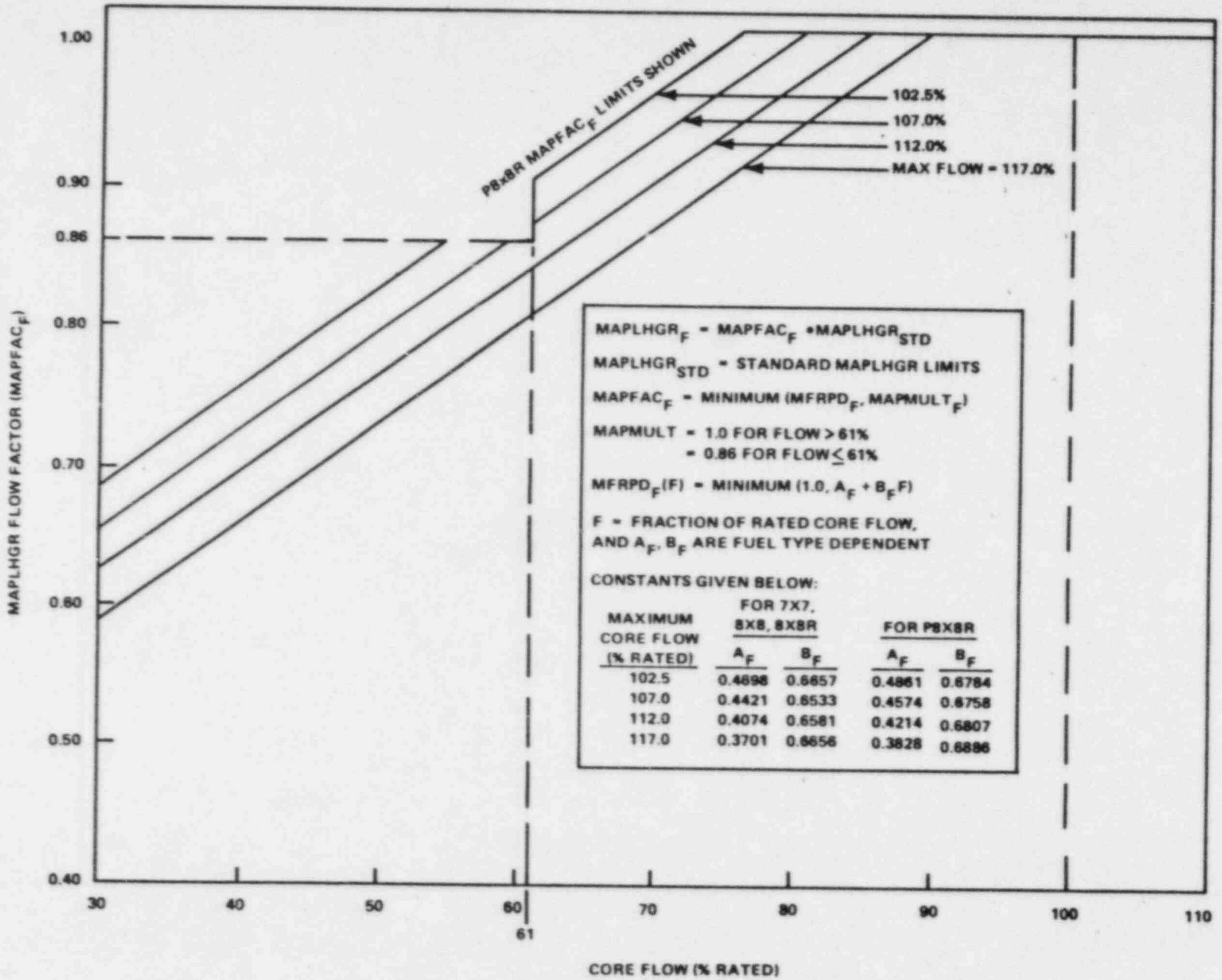


FIGURE 3.2.1-12 MAPFAC$_F$

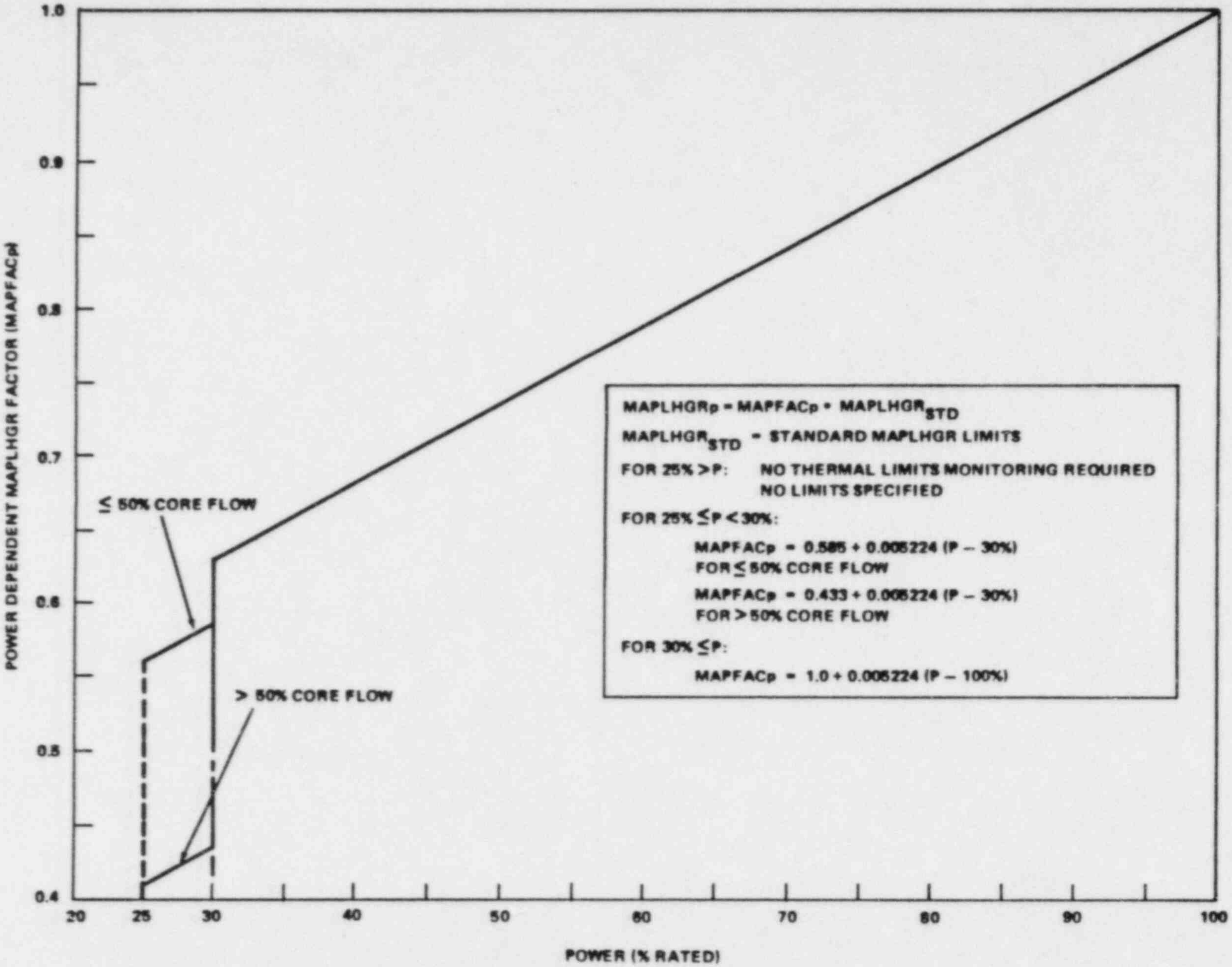


FIGURE 3.2.1-13 MAPFACp

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.

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

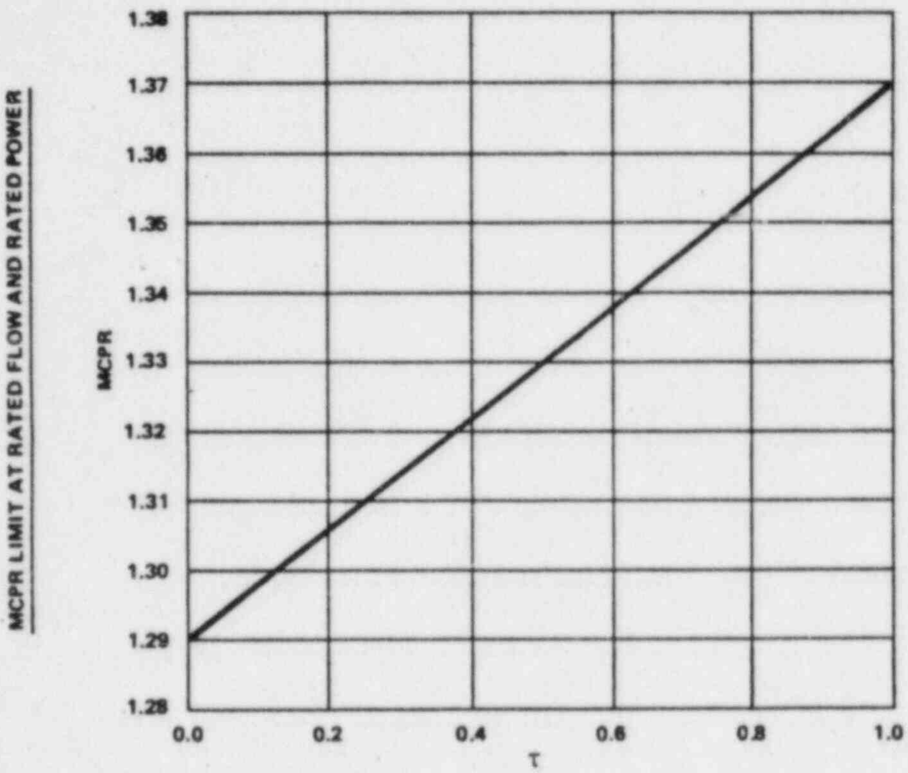
b. τ as defined in Specification 3.2.3; the determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2.

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

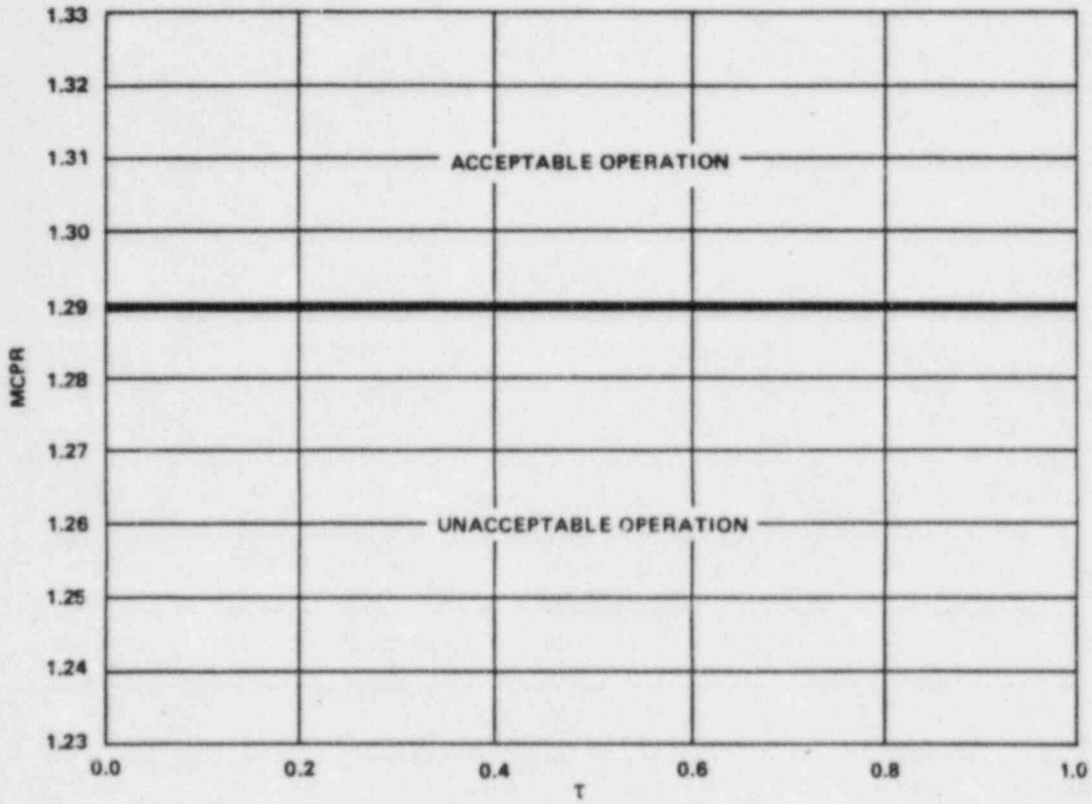


FIGURE 3.2.3-2
MCPR LIMIT FOR 7X7 FUEL
AT RATED FLOW AND RATED POWER

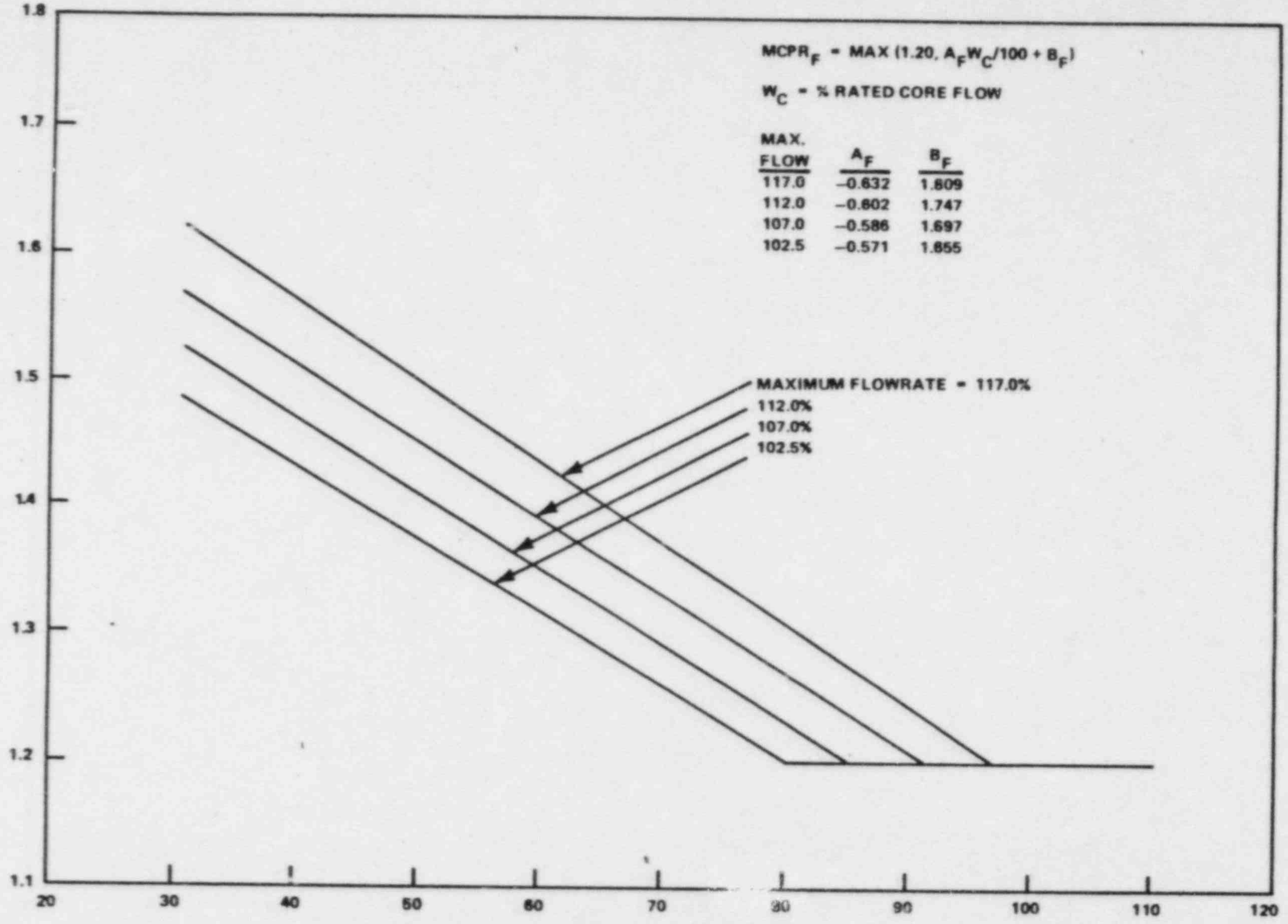


FIGURE 3.2.3-3 $MCPR_F$

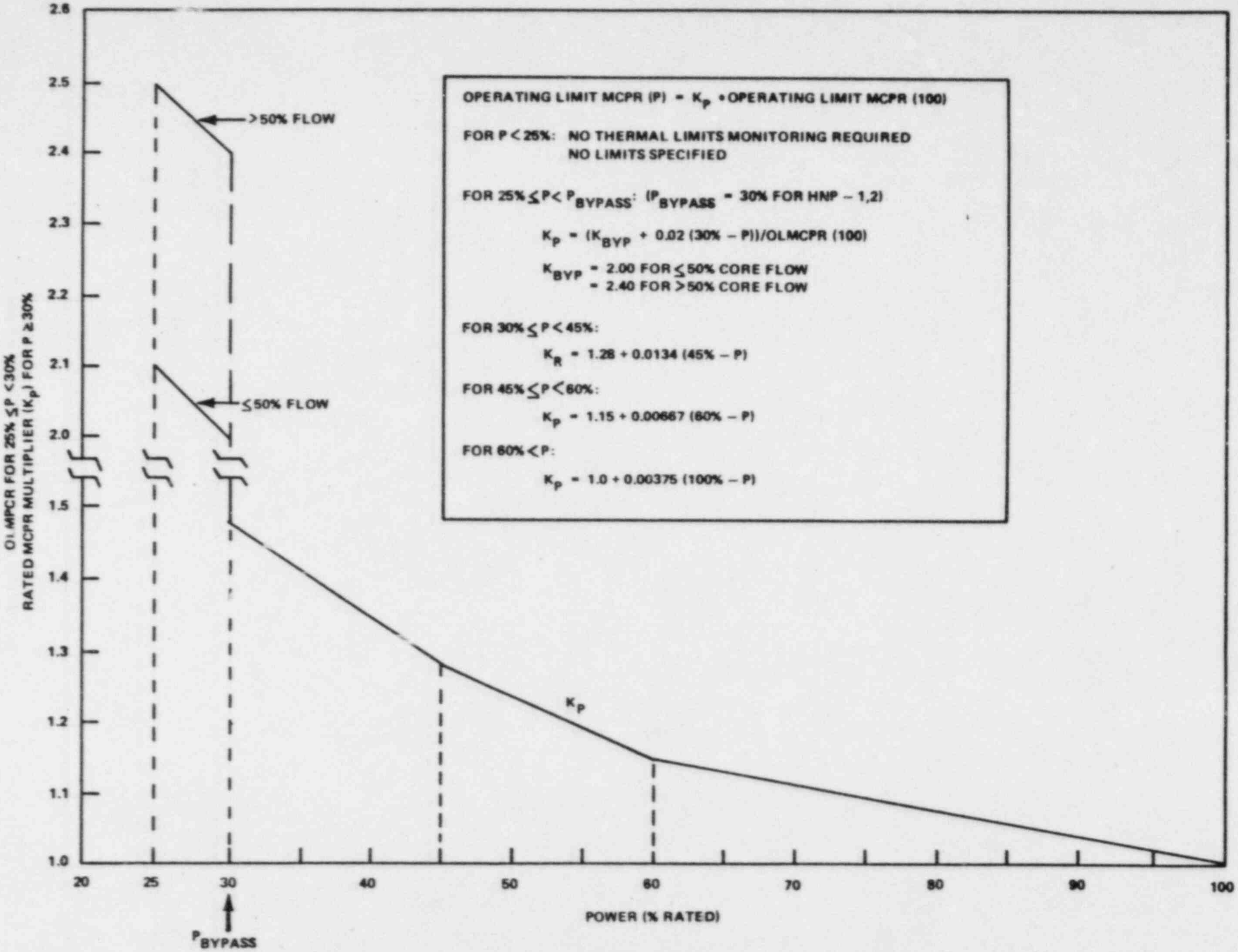


FIGURE 3.2.3-4 Kp

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.

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 ft^2

b. Small Breaks .... 1.0, 0.9, 0.4 and 0.07 ft^2

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.

POWER DISTRIBUTION LIMITS

BASES

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 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.[4]

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).

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.

Georgia Power

ENCLOSURE 4

TECHNICAL SPECIFICATIONS REVISIONS ROR
RWM AND RSCS OPERATION, FUEL STORAGE REQUIREMENTS,
FUEL ASSEMBLY DESIGN, MAPLHGR LIMITS, EDITORIAL CHANGES
PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
REFERENCED DOCUMENTS

The enclosure provides two letters referenced in Enclosure 1 which have not been previously submitted to the Nuclear Regulatory Commission.

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GENERAL ELECTRIC

NUCLEAR ENERGY BUSINESS OPERATIONS
NUCLEAR TECHNOLOGIES & FUEL DIVISION
175 Curtner Avenue
San Jose, CA 95125
M/C 174

January 10, 1986
CJP:86-005

cc: J. S. Charnley
L. K. Mathews
C. J. Paone
G. D. Plotycia
K. G. Turnage
P. VanDiemen

Mr. W. R. Mertz
Southern Company Services
P. O. Box 2625
Birmingham, AL 35202

SUBJECT: High Density Spent Fuel Storage Racks at Plant Hatch

REFERENCE: "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-7, August, 1985 (GESTAR)

Dear Mr. Mertz:

The following is in response to your recent request for information.

Currently, General Electric has a single high density spent fuel storage rack design. A criticality analysis, with design basis fuel, has produced the fuel bundle k-infinity storage criteria described in GESTAR, which are applicable to all sized modules of the GE-designed high density spent fuel storage rack. Different sized modules of the high density rack are provided for optimizing fuel storage for various pool sizes and configurations. All of the modules, however, have the same storage pitch and tube design of neutron absorbing material and, therefore, are bounded by the GESTAR k-infinity limit.

If there are any further questions, please call.

Very truly yours,

J. P. Nibert

J. P. Nibert
Fuel Project Engineer
Hatch 1, 2
(408) 925-5345

JPN:jn

86010PR00001

GENERAL ELECTRIC

NUCLEAR ENERGY BUSINESS OPERATIONS
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March 31, 1986
CJP:86-076

cc: J. S. Charnley
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D. C. Serell
K. G. Turnage

Mr. L. K. Mathews
Manager, Nuclear Core Analysis
Southern Company Services
P.O. Box 2625
Birmingham, AL 35202

SUBJECT: Hatch 2 MAPLHGR Limits for Several Fuel Types

Dear Mr. Mathews:

At the request of Southern Company Services, GE has performed an analysis of the MAPLHGR limits for several additional fuel bundle types for Hatch 2. In view of the results, GE's current plan is to provide the following MAPLHGR Tech. Spec. limits in the supplemental reload licensing report for Hatch 2 Cycle 7:

1) Add results for the P8DRB283 80 mil bundle to supplement the 100 mil results.

2) Revise the P8DRB284H MAPLHGR's to bound both 80 and 100 mil results.

3) Add clarifying notes to specify that the P8DRB265H and P8DRB299 results bound 80 and 100 mil results.

Please refer to the attached tables of MAPLHGR limits for these fuel types.

Additionally, MAPLHGR's and PCT results developed for Hatch 2 are applicable to Hatch 1 since the Hatch 2 LOCA analysis is slightly more conservative than that for Hatch 1.

If there are any questions, please call.

Very truly yours,

J. P. Nibert

J. P. Nibert
Fuel Project Engineer
Hatch 1, 2
(408) 925-5345

JPN:jn

Attachment

MAPLHGR TABLE FOR BUNDLE TYPE: P8DRB299*

EXPOSURE (GWD/ST)	EXPOSURE (GWD/MT)	MAPLHGR (KW/FT)	PCT (DEG-F)	LOCAL OXIDATION (FRACTION)
0.20	0.22	10.90	2072.	0.023
1.0	1.1	11.00	2074.	0.023
5.0	5.5	11.50	2119.	0.026
10.	11.	12.10	2199.	0.033
15.	17.	12.10	2198.	0.033
20.	22.	12.00	2197.	0.033
25.	28.	11.50	2152.	0.029
30.	33.	11.00	2056.	0.021
35.	39.	10.30	1972.	0.030
40.	44.	9.70	1855.	0.010
45.	49.	9.00	1781.	0.007

* 80 mil and 100 mil channel thickness

(CONTINUED)

General Electric Company
Nuclear Energy Business Operations
Nuclear Technologies and Fuel Division

MAPLHGR TABLE FOR BUNDLE TYPE: P8DRB283*

EXPOSURE (GWD/ST)	(GWD/MT)	MAPLHGR (KW/FT)	PCT (DEG-F)	LOCAL OXIDATION (FRACTION)
0.20	0.22	11.30	2133.	0.029
1.0	1.1	11.40	2134.	0.028
5.0	5.5	11.90	2185.	0.033
10.	11.	12.10	2195.	0.033
15.	17.	12.10	2199.	0.033
20.	22.	11.90	2184.	0.032
25.	28.	11.30	2112.	0.025
30.	33.	11.10	2061.	0.021
35.	39.	10.50	1981.	0.030
40.	44.	9.80	1808.	0.017
45.	49.	9.20	1788.	0.008

* 100 mil channel thickness

(CONTINUED)

MAPLHGR TABLE FOR BUNDLE TYPE: P8DRB265H*

EXPOSURE (GWD/ST)	(GWD/MT)	MAPLHGR (KW/FT)	PCT (DEG-F)	LOCAL OXIDATION (FRACTION)
0.20	0.22	11.50	2148.	0.030
1.0	1.1	11.60	2157.	0.030
5.0	5.5	11.90	2184.	0.032
10.	11.	12.10	2198.	0.033
15.	17.	12.10	2200.	0.033
20.	22.	11.90	2188.	0.032
25.	28.	11.30	2113.	0.025
30.	33.	10.70	2027.	0.019
35.	39.	10.20	1939.	0.014
40.	44.	9.60	1840.	0.009
45.	49.	8.90	1756.	0.007

* 80 mil and 100 mil channel thickness

(CONTINUED)

General Electric Company
Nuclear Energy Business Operations
Nuclear Technologies and Fuel Division

MAPLHGR TABLE FOR BUNDLE TYPE: P8DRB284H*

EXPOSURE (GWD/ST)	(GWD/MT)	MAPLHGR (KW/FT)	PCT (DEG-F)	LOCAL OXIDATION (FRACTION)
0.20	0.22	11.20	2111.	0.026
1.0	1.1	11.20	2106.	0.026
5.0	5.5	11.70	2152.	0.029
10.	11.	12.00	2189.	0.032
15.	17.	12.00	2195.	0.033
20.	22.	11.70	2181.	0.032
25.	28.	11.00	2089.	0.024
30.	33.	10.30	1987.	0.016
35.	39.	9.70	1887.	0.011
40.	44.	9.10	1797.	0.008
45.	49.	8.40	1716.	0.006

* 80 mil and 100 mil channel thickness

(CONTINUED)

General Electric Company
Nuclear Energy Business Operations
Nuclear Technologies and Fuel Division

MAPLHGR TABLE FOR BUNDLE TYPE: P8DRB283*

EXPOSURE (GWD/ST)	(GWD/MT)	MAPLHGR (KW/FT)	PCT (DEG-F)	LOCAL OXIDATION (FRACTION)
0.20	0.22	11.20	2116.	0.027
1.0	1.1	11.20	2113.	0.026
5.0	5.5	11.80	2171.	0.031
10.	11.	12.00	2186.	0.032
15.	17.	12.10	2199.	0.033
20.	22.	11.90	2190.	0.032
25.	28.	11.40	2124.	0.026
30.	33.	10.80	2038.	0.019
35.	39.	10.30	1942.	0.014
40.	44.	9.60	1845.	0.010
45.	49.	8.90	1763.	0.007

* 80 mil channel thickness

(END)