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) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figure 3.2.1-1 thru 3.2.1-8.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figures 3.2.1-1 thru 3.2.1-8, initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

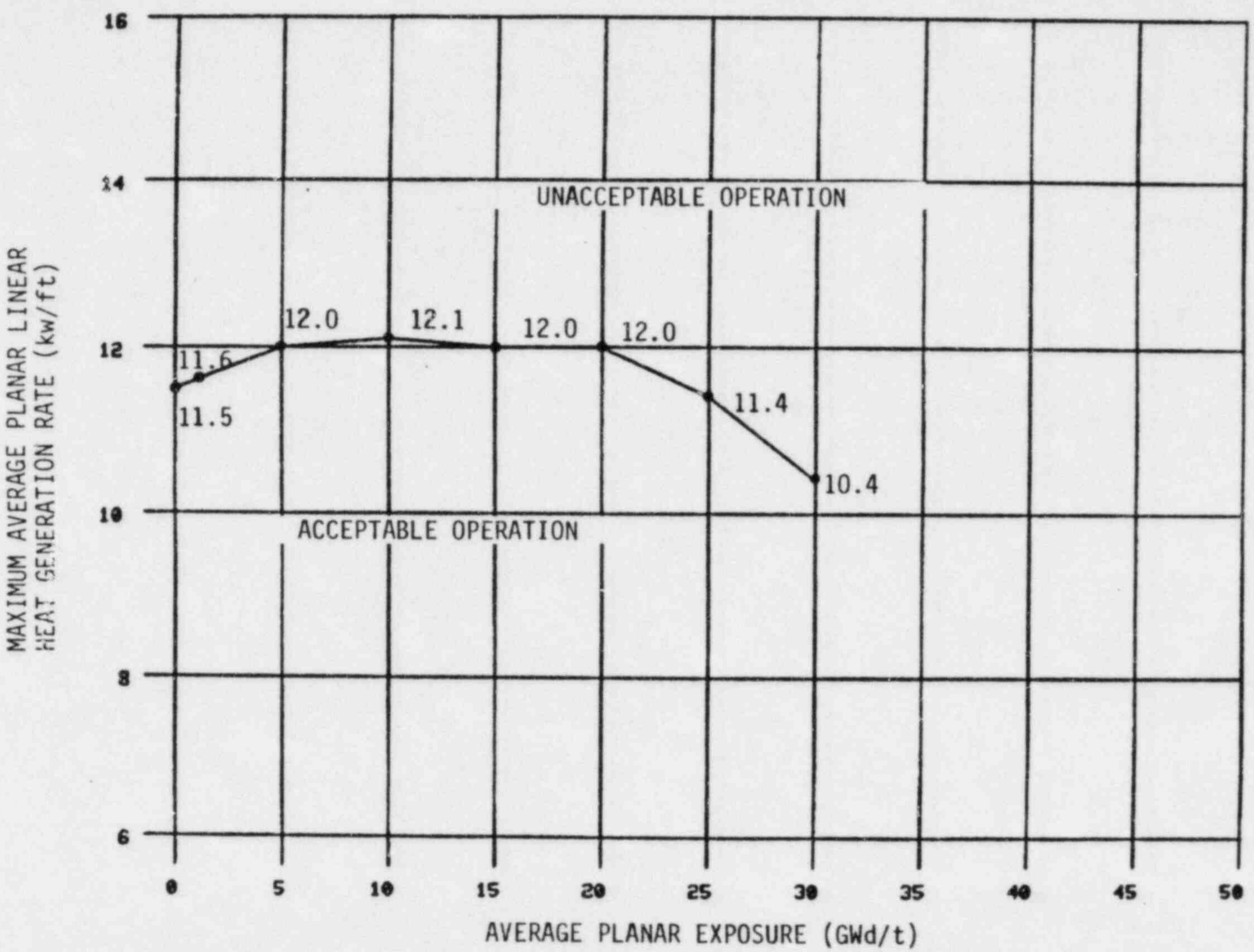
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the applicable limit determined from Figures 3.2.1-1 through 3.2.1-8:

a. At least once per 24 hours,

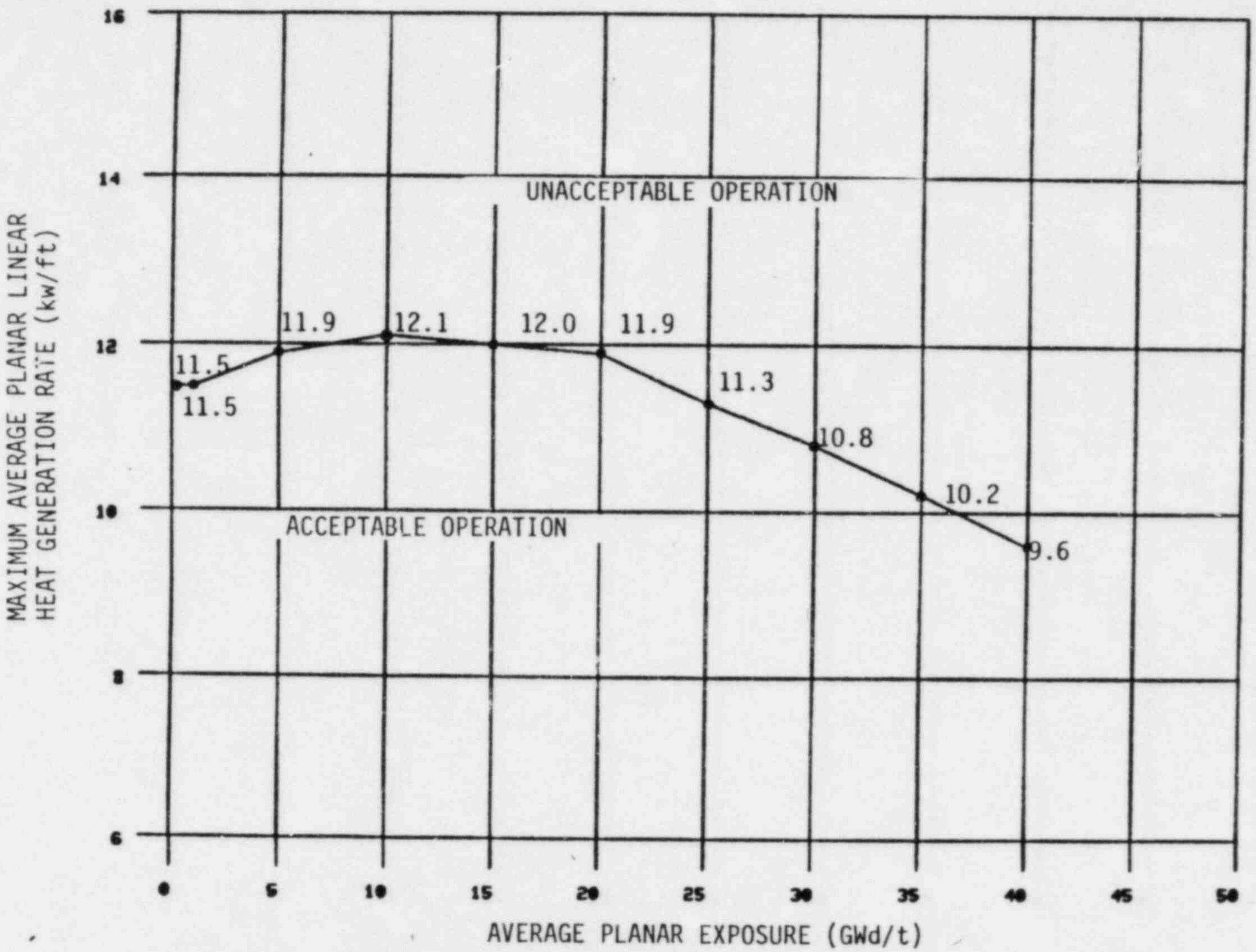
b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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



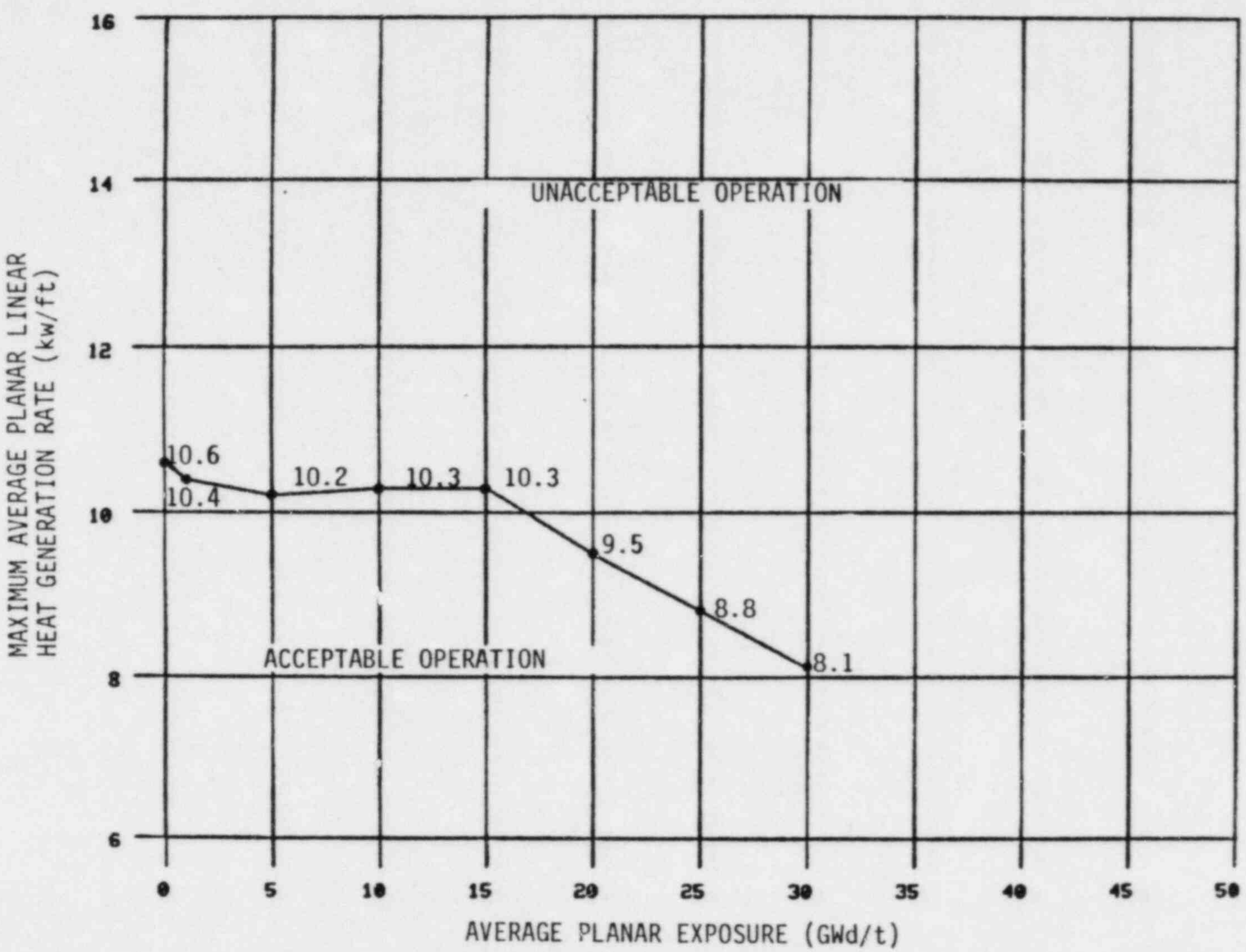
FUEL TYPE 8DIB175(8DRL183)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-1



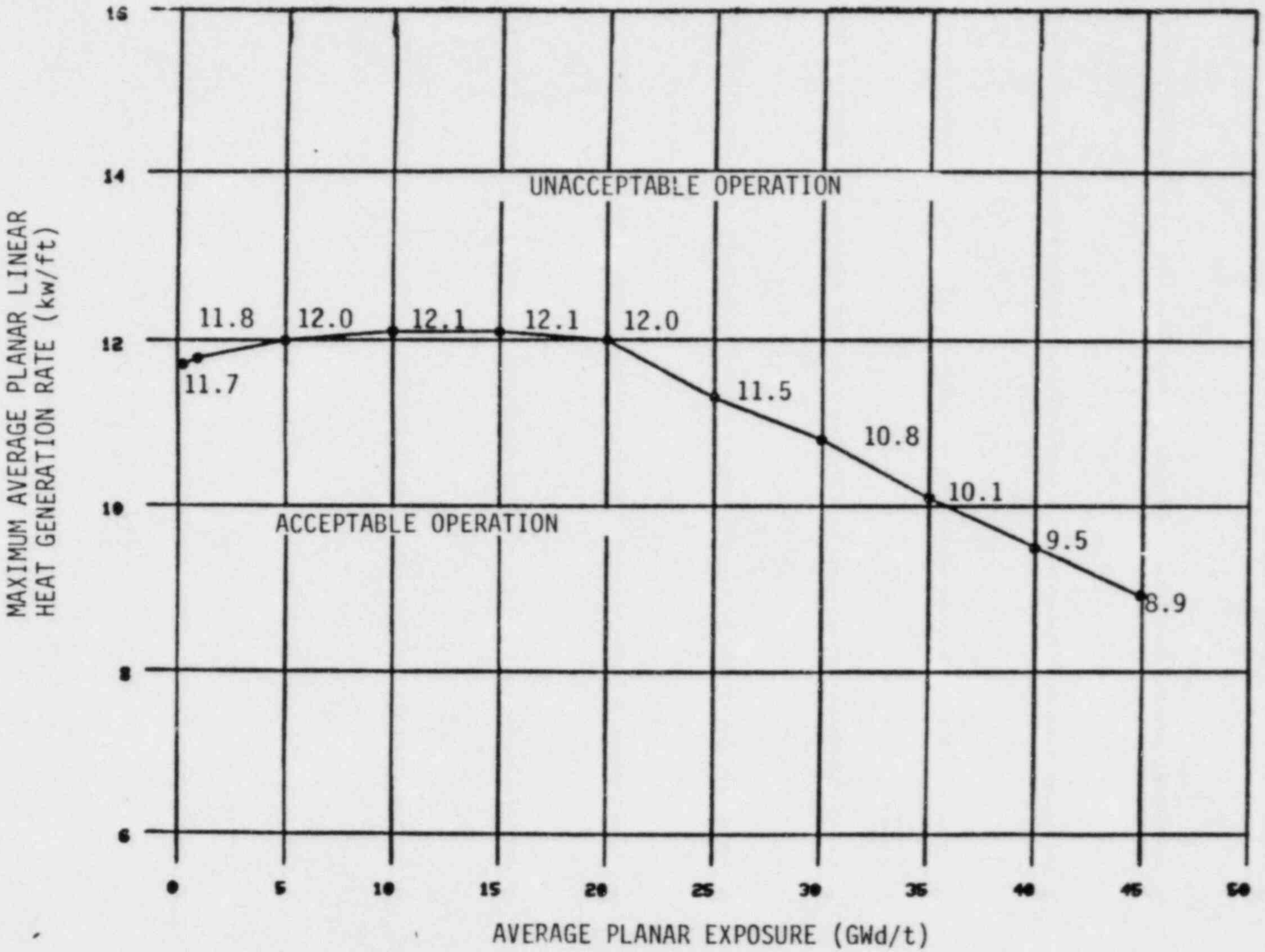
FUEL TYPE 8DIB221(8DRL233)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-2



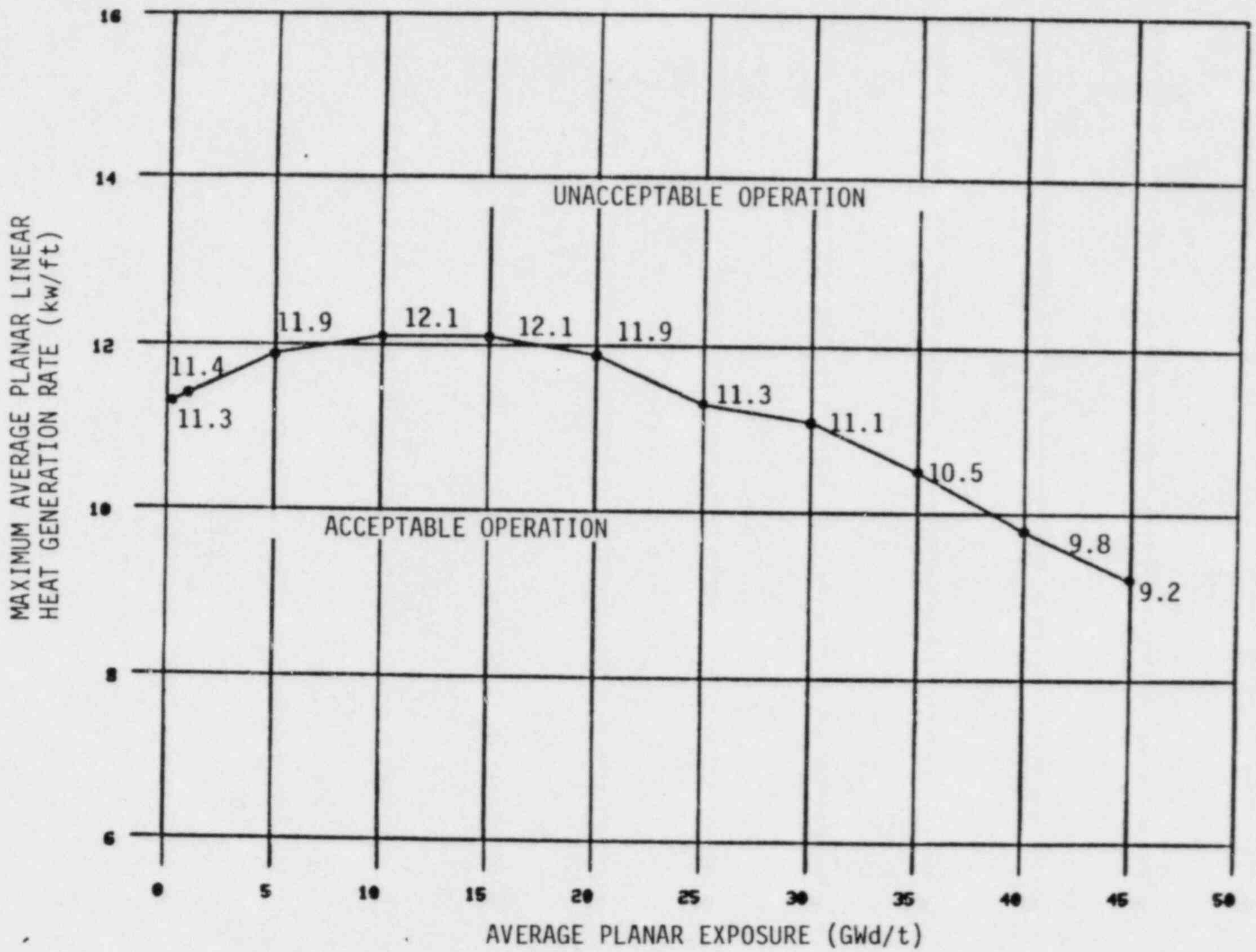
FUEL TYPE IE 711-00GD-100 MIL
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-3



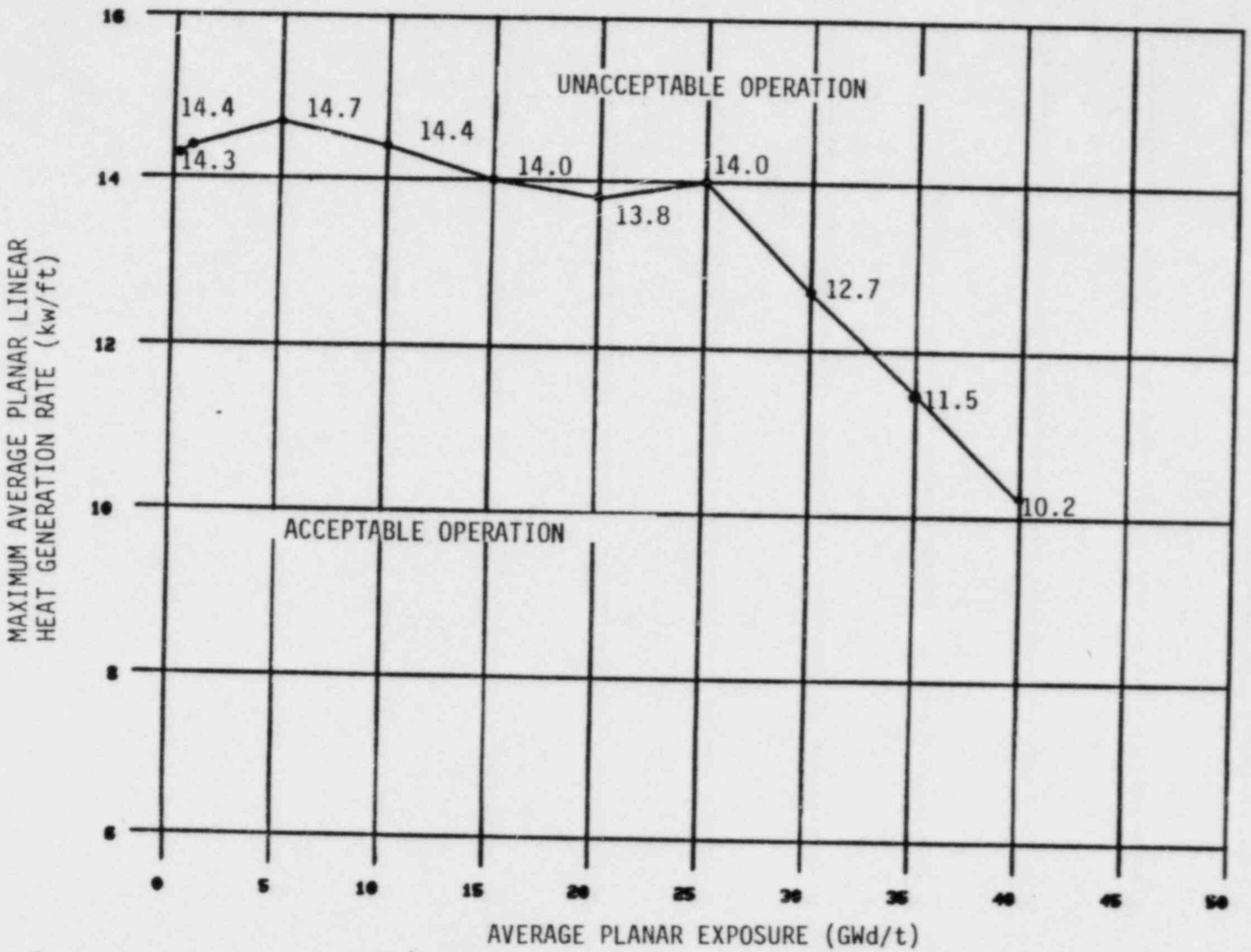
FUEL TYPE P8DRB284LA
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-4



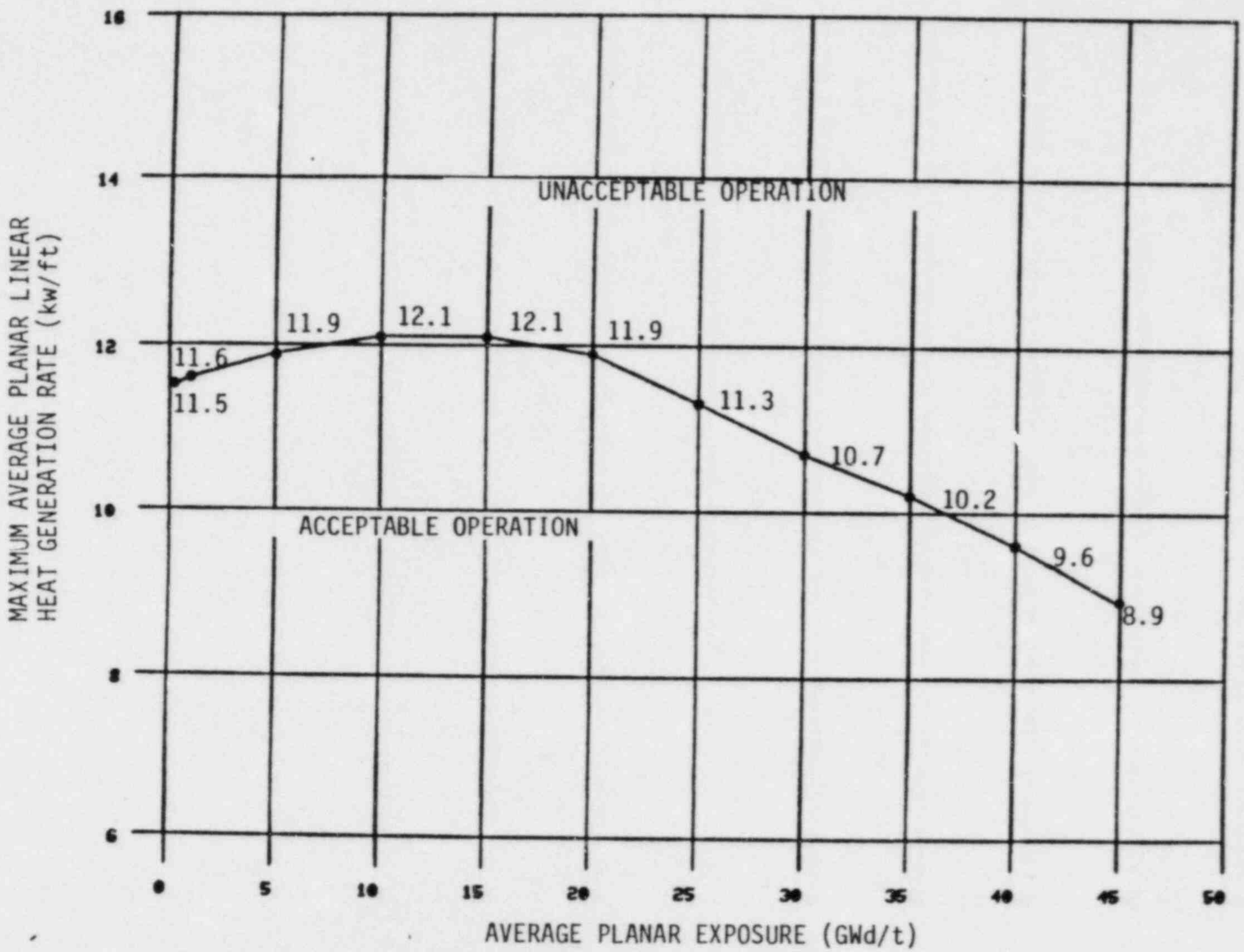
FUEL TYPE P8DRB283
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-5



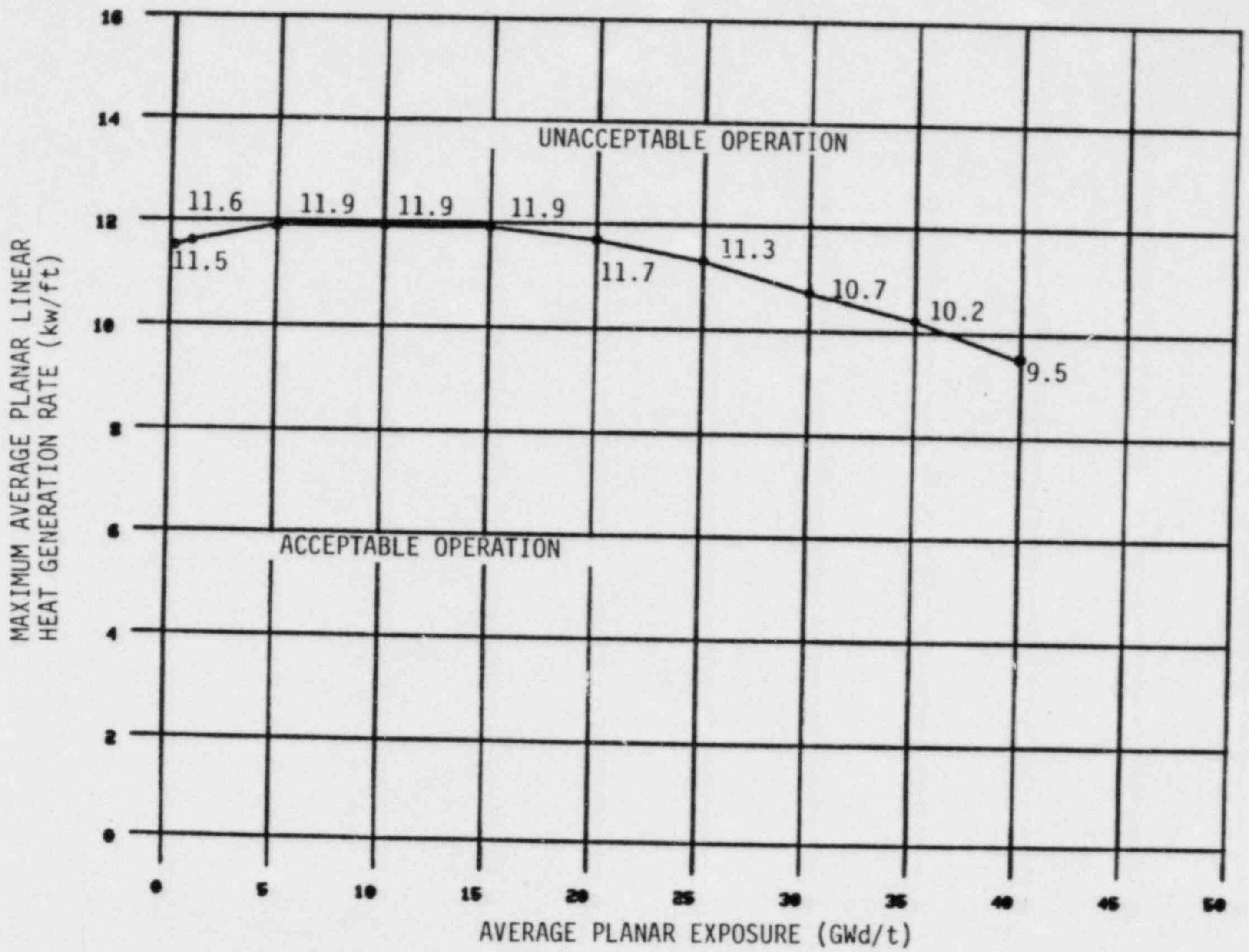
FUEL TYPE HATCH-1 I.C. 1,2,3 (7X7)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-6

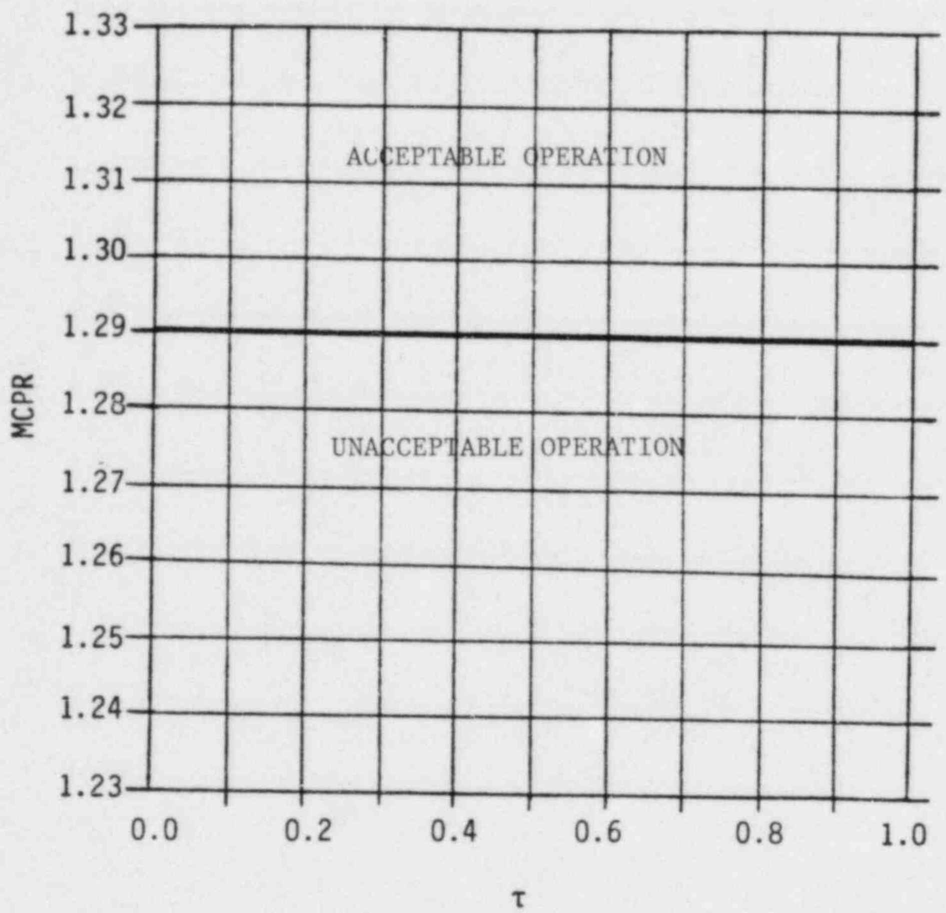


FUEL TYPE P8DRB265H
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION
RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-7

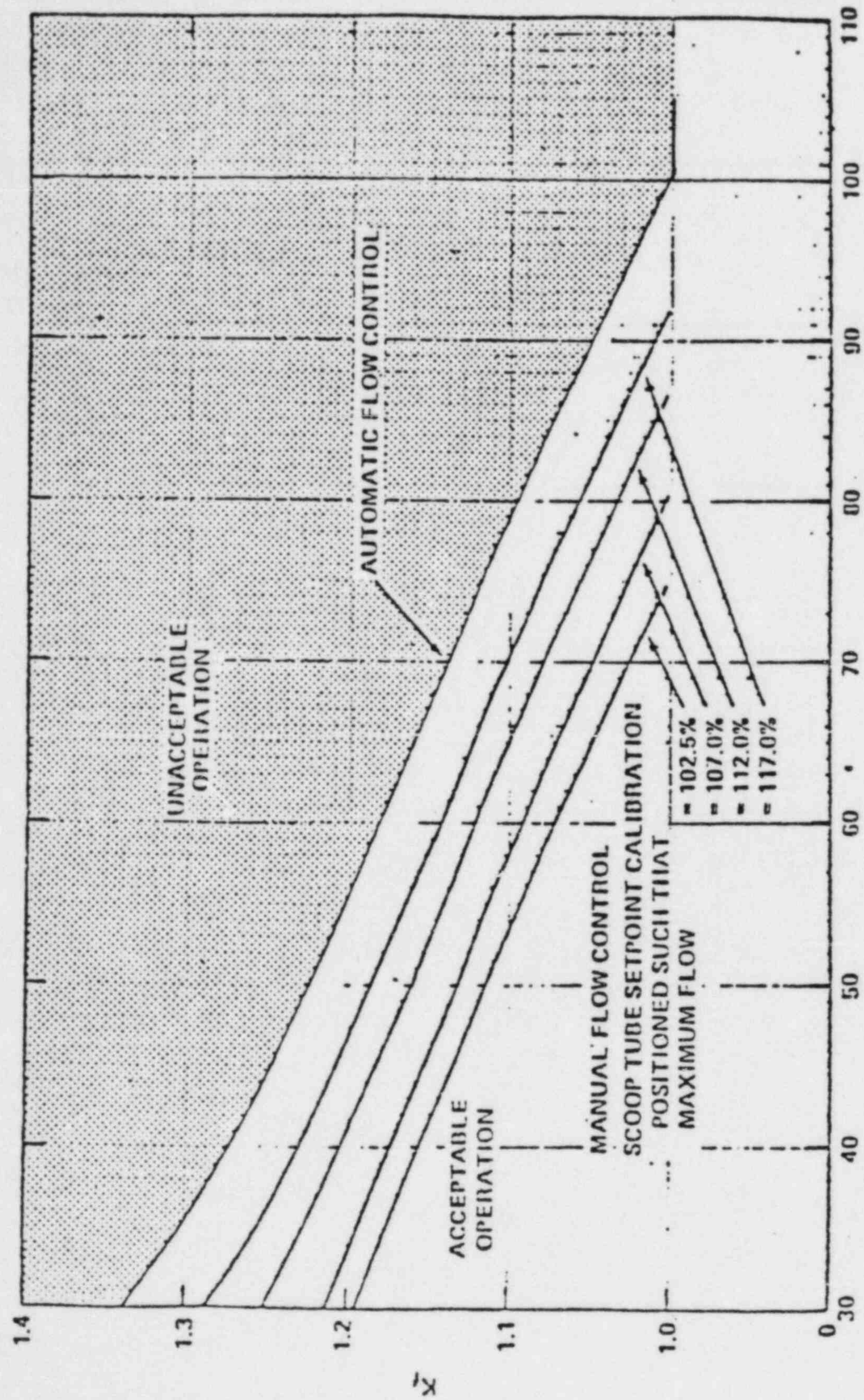


FUEL TYPE 8DRB265H
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-8



MCPR LIMIT FOR 7X7 FUEL AT RATED FLOW

FIGURE 3.2.3-3



Core Flow, % of Rated Core Flow

K_F Factor

FIGURE 3.2.3-4

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 fuel or 18.0 Kw/ft for 7X7 fuel.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$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.

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 initial core shall contain 560 fuel assemblies with each fuel assembly containing 62 fuel rods and 2 water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 3341 grams uranium. The initial core loading shall have a maximum average enrichment of 1.87 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum average enrichment of 2.90 weight percent U-235. 7X7 fuel containing 49 fuel rods and no water rods may also be inserted.

BASES

3/4.1.3 CONTROL RODS

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

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

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

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

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than 1.07 during the limiting power transient analyzed in Section 15 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications provide the required protection and MCPR remains greater than 1.07. The occurrence of scram times longer than those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram

3/4.2 POWER DISTRIBUTION

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.

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 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 limitiation 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 presented in bases Table B 3.2.1-1.

MINIMUM CRITICAL POWER RATION (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 K_f factor is to define operating limits at other than related flow conditions. At less than 100% of rated flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of Specification 3.2.3 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated should the most limiting transient occur at less than rated flow.

The K_f factor values shown in Figure 3.2.3-4 were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors wre derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that the maximum flow rate, as limited by the pump scoop tube set point and the corresponding THERMAL POWER along the rated 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 rated flow control line corresponding to different core flow. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

MINIMUM CRITICAL POWER RATION (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and rated flow.

The K_f factors shown in Figure 3.2.3-4 are conservative for the General Electric Plant operation because the operating limit MCPRs of Specification 3.2.3 are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial startup testing of the plant, an MCPR evaluation will be made at 25% of RATED THERMAL POWER with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

The LHGR specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

ATTACHMENT 3

NRC DOCKET 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
PAGE-BY-PAGE DESCRIPTION OF CHANGES

3/4 2-1	-	Add new Figure numbers
3/4 2-2	-	No Change: Better quality graph
3/4 2-3	-	Extend MAPLHGR curve to higher exposure
3/4 2-4a	-	Change page number
3/4 2-4b	-	Extend MAPLHGR curve to higher exposure; change page number
3/4 2-4c	-	Extend MAPLHGR curve to higher exposure; change page number
3/4 2-4d	-	Add Unit 1 7x7 MAPLHGR curve
3/4 2-4e	-	Add new Unit 2 fuel MAPLHGR curve
3/4 2-4f	-	Add Unit 1 reconstituted fuel MAPLHGR curve
3/4 2-6	-	Add new figure number; reflect new K_f figure number; add reference to 7x7 fuel
3/4 2-7b	-	Add MCPR limit curve for 7x7 fuel
3/4 2-7c	-	Change figure number and page number
3/4 2-8	-	Add LHGR for 7x7 fuel
3/4 5-1	-	Change "reactor core" to "initial core"; correct "Zircaloy-4 to "Zircaloy-2"; add sentence about 7x7 fuel
B 3/4 1-2	-	Reflect new safety limit of 1.07
B 3/4 2-1	-	Condense reference to APLHGR figures
B 3/4 2-4	-	Reflect change in K_f curve figure number
B 3/4 2-5	-	Reflect change in K_f curve figure number

ATTACHMENT 4

NRC DOCKET 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
REFERENCE DOCUMENTATION

Y1003J01A57, "Supplemental Reload Licensing Submittal for Edwin I. Hatch Nuclear Plant Unit 2, Reload 3 (Cycle 4)", January 1983.

LMQ:83-018, "Hatch 2 ECCS MAPLHGR Calculations for Hatch 1 Initial Core and Reconstituted Hatch 1 Reload 2 Fuel", February 22, 1983

LMQ:83-022, "Hatch 2 Cycle 4 OLMCPR", February 24, 1983.

GENERAL ELECTRIC

NUCLEAR FUEL AND SPECIAL PROJECTS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125

February 24, 1983
LMQ:83-022

cc: B. E. Hunt
L. K. Mathews
H. C. Nix

Mr. R. D. Baker
Georgia Power Company
P.O. Box 4545
Atlanta, GA 30302

Subject: Hatch 2 Cycle 4 OLMCPR

References:
1) "Supplemental Reload Licensing Submittal for Edwin I. Hatch Nuclear Plant Unit 2 Reload 3 (Cycle 4)", Y1003J01A57, Rev. 0, 1/83

2) "General Electric Boiling Water Reactor Load Line Limit Analysis for Edwin I. Hatch Nuclear Plant Unit 2", NEDO-24295, 10/80, as amended

3) "Safety Review of Hatch Nuclear Power Station Unit No. 2 at Core Flow Conditions above Rated Core Flow Throughout Cycle 2," NEDO-24292, Rev, 2 10/81

Dear Mr. Baker:

The operating limit minimum critical power ratios (OLMCPR's) documented in Reference 1 for Hatch 2 Cycle 4 are based on operation at rated flow and final feedwater temperature and assume a full-arc turbine control valve configuration in the transient analyses. These results bound both operation above the 100% power/100% flow load line at rated final feedwater temperature and a partial-arc turbine control valve configuration.

The OLMCPR's in Reference 1 are also sufficiently conservative to bound operation in Cycle 4 before nominal end-of-cycle at increased core flow (up to 105% of rated) and/or final feedwater temperature reduction (up to 63 F ΔT, on or below the 100% power/100% flow load line) conditions on an intermittent basis, full-arc or partial-arc turbine control valves.

GENERAL ELECTRIC

This conclusion is based on the results of analyses performed by General Electric for Cycle 4, final documentation of which will be included in a revision to Reference 1. If you have any questions on this subject, please do not hesitate to contact us.

Very truly yours,

Louis M. Quintana

L. M. Quintana
Fuel Project Specialist
Hatch 1 & 2
M/C 174; (408) 925-2026

pm

CONCURRENCE: C L Hilf
C. L. Hilf, S&LO

GENERAL ELECTRIC

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125

NUCLEAR FUEL AND SPECIAL PROJECTS DIVISION

February 22, 1983
LMQ:83-018

cc: R. D. Baker
B. E. Hunt
H. C. Nix
R. T. Schellinger

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

Subject: Hatch 2 ECCS MAPLHGR Calculations for Hatch 1 Initial Core and Reconstituted Hatch 1 Reload 2 Fuel

Dear Mr. Mathews:

Per Mr. K. S. Folk's verbal request, General Electric Company has performed an analysis of the ECCS MAPLHGR limits for the Hatch 1 initial core and reconstituted Hatch 1 Reload 2 fuel bundles when inserted in Hatch 2. This analysis was required because, although the Hatch 1 and Hatch 2 plants are similar, the Hatch 2 plant ECCS analysis is slightly more severe due to a different limiting break size and associated core uncovery and reflood times. The results of this analysis are documented in the attached tables.

The three Hatch 1 7x7 initial core fuel types' MAPLHGR's and PCT's are represented in the tables as one fuel type since bounding calculations were made to cover all three. These calcultions were made for 80 mil channels only since local peaking data for these 7x7 lattice with 100 mil channels are not available. However, the trend observed from the 8DRB265-6G3.0 results for 80 versus 100 mil channels shows less than a 0.1 Kw/ft MAPLHGR difference between the two, with the 80 mil channel results being slightly worse. It is concluded that the MAPLHGR limits for the 7x7 bundles based on 80 mil channels are applicable to 100 mil channels as well, should the thicker channels be required.

The un-reconstituted local peaking factors for the 8DRB265-6G3.0 bundle were used for this analysis since the reconstituted bundle has slightly better ECCS performance. (The reconstituted bundle has better thermal conductivity and local peaking characteristics than the original bundle.) Although the results are not very sensitive to 80 or 100 mil channels, the reported 8DRB265-6G3.0 MAPLHGR results bound the use of either channel.

GENERAL ELECTRIC

If you have any questions concerning this subject, please do not hesitate to contact us.

Very truly yours,

Louis M. Quintana

L. M. Quintana
F el Project Specialist
Hatch 1 & 2
M/C 174; (408) 925-2026

pm
Enclosures

G E N E R A L   E L E C T R I C

NUCLEAR ENERGY ENGINEERING DIVISION

HATCH-2 R3/C4 ECCS REV.1

I MAPLHGR TABLE FOR BUNDLE TYPE: HATCH-1 I.C. TYPE 1,2,3

EXPOSURE (GWD/ST)	(GWD/MT)	MAPLHGR (KW/FT)	PCT (DEG-F)	LOCAL OXIDATION (FRACTION)
.20	.22	14.30	2198.	0.036
1.0	1.1	14.40	2198.	0.035
5.0	5.5	14.70	2197.	0.032
10.	11.	14.40	2199.	0.031
15.	17.	14.00	2198.	0.057
20.	22.	13.80	2199.	0.057
25.	28.	14.00	2198.	0.052
30.	33.	12.70	2013.	0.017
35.	39.	11.50	1886.	0.010
40.	44.	10.20	1729.	0.006

II MAPLHGR TABLE FOR BUNDLE TYPE: 8DRB265-6G3-80M/100M

EXPOSURE (GWD/ST)	(GWD/MT)	MAPLHGR (KW/FT)	PCT (DEG-F)	LOCAL OXIDATION (FRACTION)
.20	.22	11.50	2164.	0.031
1.0	1.1	11.60	2167.	0.031
5.0	5.5	11.90	2199.	0.034
10.	11.	11.90	2200.	0.033
15.	17.	11.90	2200.	0.034
20.	22.	11.70	2198.	0.034
25.	28.	11.30	2147.	0.029
30.	33.	10.70	2067.	0.022
35.	39.	10.20	1994.	0.017
40.	44.	9.50	1904.	0.012