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

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 33
License No. DPR-16

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

   A. The application for amendment by Jersey Central Power & Light Company (the licensee) dated May 30, 1978, as supplemented by letters dated June 6, 1978 and October 3, 1978, 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 33 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: November 11, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 33

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3.10-1	3.10-1
3.10-2	3.10-2
3.10-3	3.10-3 *
3.10-4	3.10-4
3.10-5	3.10-5
3.10-6	3.10-6
3.10-7	3.10-7
3.10-8	3.10-8
---	3.10-9
---	3.10-10
4.10-1	4.10-1

*There is no change on this page. The contents of the page have merely been repositioned.

3.10 CORE LIMITS

Applicability: Applies to core conditions required to meet the Final Acceptance Criteria for Emergency Core Cooling Performance.

Objective: To assure conformance to the peak clad temperature limitations during a postulated loss-of-coolant accident as specified in 10 CFR 50.46 (January 4, 1974) and to assure conformance to the 17.2 KW/ft. (for 7 x 7 fuel) and 14.5 KW/ft. (for 8 x 8 fuel) operating limits for local linear heat generation rate.

Specification: A. Average Planar LHGR

During power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location shall not exceed the product of the maximum average planar LHGR (MAPLHGR) limit shown in Figure 3.10-1 and the axial MAPLHGR multiplier in Figure 3.10-2. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

B. Local LHGR

During power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly, at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR \leq LHGR_d \left[1 - \frac{(\Delta P)}{P} \max \left(\frac{L}{LT}\right)\right]$$

Where: $LHGR_d$ = Limiting LHGR

$\frac{\Delta P}{P}$ = Maximum Power Spiking Penalty

LT = Total Core Length - 144 inches

L = Axial position above bottom of core

and

Fuel Type	$LHGR_d$	$\Delta P/P$
II	17.2	.032
IIIE	17.1	.046
IIIF	17.1	.033
V	14.5	.033
VB	14.5	.039

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

C. Assembly Averaged Power Void Relationship (Applied to type II fuel only)

During power operation, the assembly average void fraction and assembly power shall be such that the following relationship is satisfied:

$$\left(\frac{1-VF}{PR \times FCP}\right) \geq B$$

Where: VF = Bundle averge void fraction
PR = Assembly radial power factor
FCP = Fractional core power (relative to 1930 MWt)
B = Power-Void limit

The limiting value of "B" for fuel type II is .365.

D. Minimum Critical Power Ratio (MCPR)

During steady state power operation, MCPR shall be greater than or equal to the following:

ARPM Status	MCPR Limit
1. If any two (2) LPRM assemblies which are input to the APRM system and are separated in distance by less than	1.64

three (3) times the control rod pitch contain a combination of (3) out of four (4) detectors located in either the A and B or C and D levels which are failed or bypassed (i.e., APRM channel or LPRM input bypassed or inoperable.

2. If any LPRM input to the APRM system at the B, C, or D level is failed or bypassed or any APRM channel is inoperable (or bypassed).	1.58
3. All B, C, and D LPRM inputs to the APRM system are operating and no APRM channels are inoperable or bypassed.	1.52

When APRM status changes due to instrument failure (APRM or LPRM input failure), the MCPR requirement for the degraded condition shall be met within a time interval of eight (8) hours, provided that the control rod block is placed in operation during this interval.

If at any time during power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded for reasons other than instrument failure, action shall be initiated to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

Basis:

The Specification for average planar LHGR assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46 (January 4, 1974) 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 are below the limits specified in 10 CFR 50.46 (January 4, 1974).

The maximum average planar LHGR shown in Figure 3.10-1 for Type II fuel is the result of LOCA analyses performed utilizing an evaluation model developed by General Electric Company in compliance with 10 CFR 50, Appendix K (1). Single failure considerations were based on the revised Oyster Creek Single Failure Analysis submitted to the Staff on July 15, 1975.

The maximum average planar LHGR shown in Figure 3.10-1 for Type IIIE, IIIF, V and VB fuel are the result of LOCA analyses performed by Exxon Nuclear Company utilizing an evaluation model developed by Exxon Nuclear Company in compliance with Appendix K to 10 CFR 50 (2). In addition, the maximum average planar LHGR shown in Figure 3.10-1 for Type V and VB fuel were analyzed with 100% of the spray cooling coefficients specified in Appendix K to 10 CFR Part 50 for 7x7 fuel. These spray heat transfer coefficients were justified in the ENC Spray Cooling Heat Transfer Test Program (3).

The effect of axial power profile peak location is evaluated for the worst break size by performing a series of fuel heat-up calculations. A set of multipliers is devised to reduce the allowable bottom skewed axial power peaks relative to center or above center peaked profiles. The major factors which lead to the lower MAPLHGR limits with bottom skewed axial power profiles are the change in canister quench time at the axial peak location and a deterioration in heat transfer during the extended downward flow period during blowdown. The MAPLHGR multiplier in Figure 3.10-2 shall only be applied to MAPLHGR determined by the evaluation model described in reference 2.

The possible effects of fuel pellet densification are: 1) creep collapse of the cladding due to axial gap formation; 2) increase in the LHGR because of pellet column shortening; 3) power spikes due to axial gap formation; and 4) changes in stored energy due to increased radial gap size.

Calculations show that clad collapse is conservatively predicted not to occur during the exposure lifetime of the fuel. Therefore, clad collapse is not considered in the analyses.

Since axial thermal expansion of the fuel pellets is greater than axial shrinkage due to densification, the analyses of peak clad temperature do not consider any change in LHGR due to pellet column shortening. Although the formation of axial gaps might produce a local power spike at one location on any one rod in a fuel assembly, the increase in local power density would be on the order of only 2% at the axial midplane. Since small local variations in power distribution have a small effect on peak clad temperature, power spikes were not considered in the analysis of loss-of-coolant accidents.

Changes in gap size affect the peak clad temperatures by their effect on pellet clad thermal conductance and fuel pellet stored energy. Treatment of this effect combined with the effects of pellet cracking, relocation and subsequent gap closure are discussed in NEDO-20181 and XN-174.

Pellet-clad thermal conductance for Type II fuel was calculated using the GEGAP III model (NEDO-20181) and Pellet-clad thermal conductance for Type IIIE, IIIF, V and VB fuel was calculated using the CAPEX model (XN-174).

The specification for local LHGR assures that the linear heat generation rate in any rod is less than the limiting linear heat generation even if fuel pellet densification is postulated. The power spike penalty specified for Type II fuel is based on the analysis presented in Section 3.2.1 of the GE Topical Report NEDM-10735 Supplement 6. The power spike penalty for Type IIIE, and IIIF fuel is based on analyses presented in Facility Change Request Nos. 4 and 5, Facility Change Request No. 6 for Type V and Amendment No. 76 for Type VB fuel. The analysis assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with 95% confidence that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

The General Electric non-jet pump BWR ECCS model (1) utilizes an empirical correlation to determine the duration of nucleate boiling heat transfer in the early period following the postulated pipe break. This correlation for time to dryout is found to be proportional to the ratio of assembly water volume to power. Dryout time is a significant parameter in determining the extent of nucleate and transition boiling heat transfer, and consequently the peak cladding temperature.

By maintaining reactor power and void fraction as specified in 3.10.C, dryout times at least as long as that used in the LOCA analysis will be assured. The limiting value of B in Specification 3.10.C was developed for core conditions of 100% power and 70% flow, the minimum flow that could be achieved without automatic plant trip (flow biased high neutron flux scram). Such a condition is never achieved during actual operation due to the neutron flux rod block and the inherent reactor power-flow relationship. The MAPLHGR results shown in Figure 3.10-1 were evaluated for 102% power and 70% flow, thus the 2% conservatism for instrument uncertainty is retained in the limiting value of B. Additional conservatism is provided by the following assumptions used in determining the B limit.

1. All heat was assumed to be removed by the active channel flow. No credit was taken for heat removal by leakage flow (10% of total flow).

2. Each fuel type was assumed to be operating at full thermal power rather than the reduced power resulting from the more limiting conditions imposed by Figure 3.10-1.

The loss of coolant accident (LOCA) analyses are performed using an initial core flow that is 70% of the rated value. The rationale for use of this value of flow is based on the possibility of achieving full power (100% rated power) at a reduced flow condition. The magnitude of the reduced flow is limited by the flow relationship for overpower scram. The low flow condition for the LOCA analysis ensures a conservative analysis because this initial condition is associated with a higher initial quality in the core relative to higher flow-lower quality conditions at full power. The high quality-low flow condition for the steady-state core operation results in rapid voiding of the core during the blowdown period of the LOCA. The rapid degradation of coolant conditions due to voiding results in a decrease in the time to boiling transition and thus degradation of heat transfer with consequent higher peak cladding temperatures. Thus, analysis of the LOCA using 70% flow and 102% power provides a conservative basis for evaluation of the peak cladding temperature and the maximum linear heat generation rate (MAPLHGR) for the reactor.

The minimum critical power ratio (MCPR) calculated for the initial conditions of the LOCA represents the thermal margin of the hot assembly to the boiling transition point. An increase in core flow from 70% would result in additional thermal margin (higher MCPR value). The conservative ECCS analysis bounds the range of permitted reactor operating conditions so long as operating MCPR's are above the values computed for the initial conditions assumed for ECCS analysis. Current plant technical specifications (3.10.D), based upon consideration of other transients, limit the reactor operation on thermal margins substantially above the assumed ECCS conditions. The assumed initial MCPR values for the ECCS analysis are 1.37 for 7x7 fuel and 1.40 for 8x8 fuel.

For transient operation up to the fuel cladding integrity safety limit, protection is provided against a MCPR of 1.34 for 8x8 fuel and 1.32 for 7x7 fuel. The actual steady-state operating power level provides margin to this limit by an amount corresponding to the maximum decrease in CPR resulting from single operator error or equipment malfunction from a steady-state level.

These resulting operating MCPR limits, combined with the transient analysis results, provide assurance that the fuel cladding integrity safety limit will not be violated during anticipated operating transients.

The APRM response is used to predict when the rod block occurs in the analysis of the rod withdrawal error transient. The transient rod position at the rod block and corresponding MCPR can be determined. The MCPR has been evaluated for different APRM responses which would result from changes in the APRM status as a consequence of bypassed APRM channel and/or failed or bypassed LPRM inputs. The results indicate that the steady state MCPR required to protect the minimum transient MCPR of 1.34 at the rod block ranges from 1.5 to 1.6 depending on the APRM system status (4).

In order to provide for a limit which is considered to be bounding to future operating cycles, the variable limits have been conservatively adjusted upward to range from 1.52 to 1.64.

The time interval of eight (8) hours to adjust the steady state MCPR to account for a degradation in the APRM status is justified on the basis of instituting a control rod block which precludes the possibility of experiencing a rod withdrawal error transient since rod withdrawal is physically prevented. This time interval is adequate to allow the operator to either increase the MCPR to the appropriate value or to upgrade the status of the APRM system while in a condition which prevents the possibility of this transient occurring.

REFERENCES

(1) Oyster Creek Nuclear Generating Station, Loss-of-Coolant Accident Analysis Reevaluation and Technical Specification Change Request No. 42, Attachment I, dated December 23, 1975.

(2) XN-75-55-(A), XN-75-55, Supplement 1-(A), XN-75-55. Supplement 2-(A), Revision 2, "Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek plant," April 1977.

(3) XN-75-36 (NP)-(A), XN-75-36 (NP) Supplement 1-(A), "Spray Cooling Heat Transfer phase 1 Test Results, ENC - 8x8 BWR Fuel 60 and 63 Active Rods, Interim Report," October 1975.

(4) Oyster Creek Nuclear Generating Station Amendment No. 76 (Supplement No. 4) Section 2.0, dated October 20, 1975.

FIGURE 3.10-1

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE

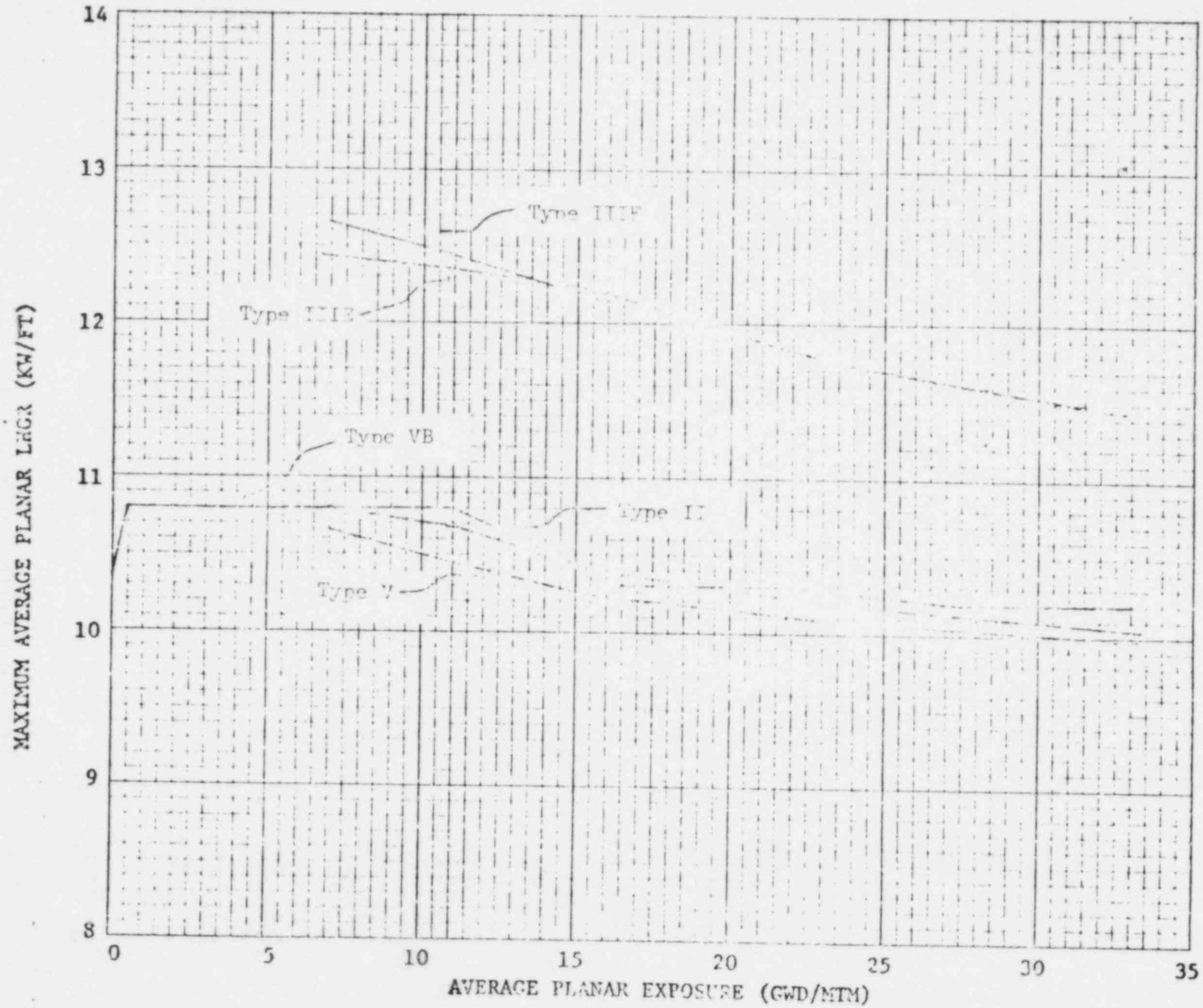

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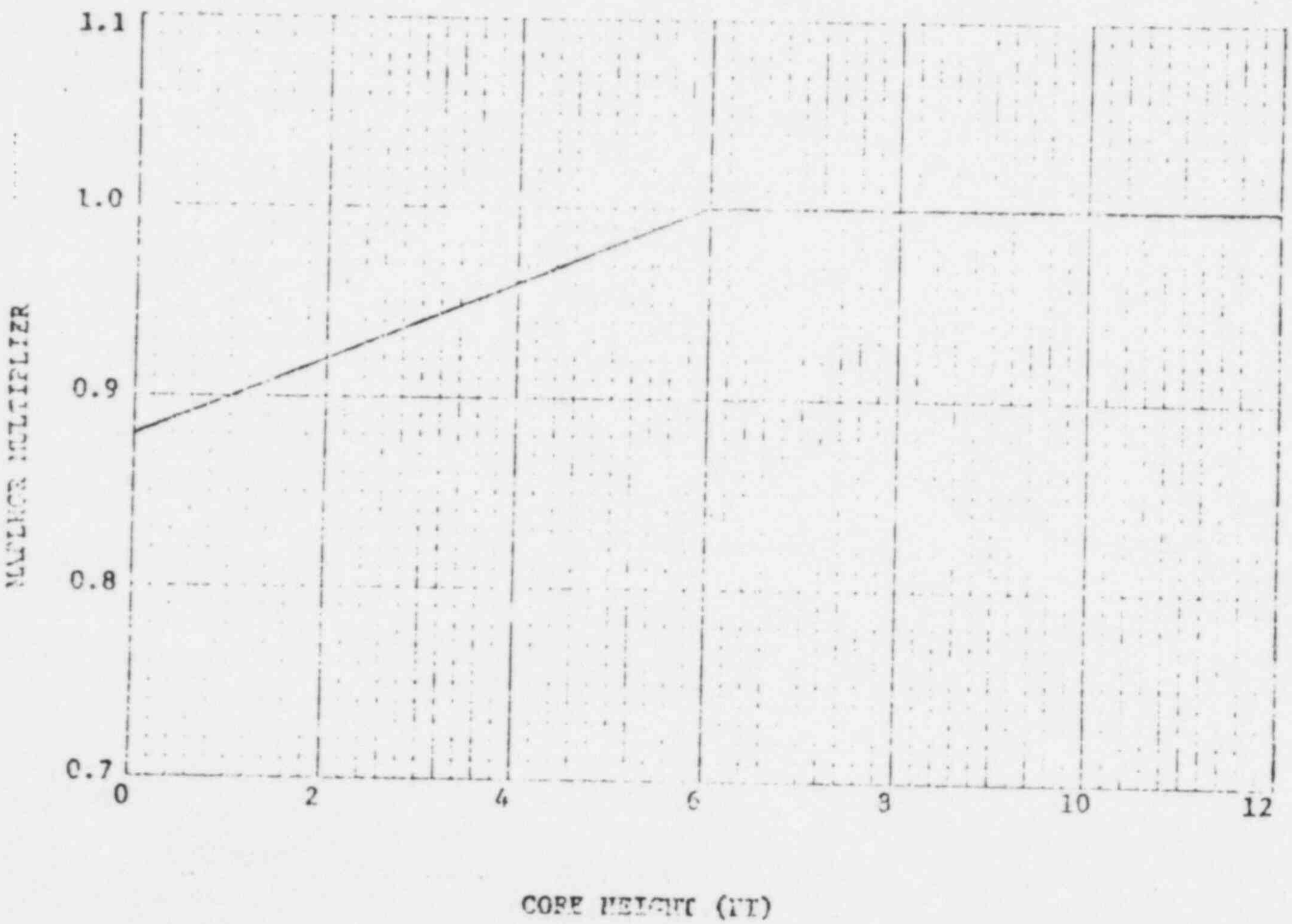
FIGURE 3.10-2

AXIAL MAPLHGR MULTIPLIER



Note : Applies to fuel types IIIE, IIIF, V, and VB.

4.10 ECCS RELATED CORE LIMITS

Applicability: Applies to the periodic measurement during power operation of core parameters related to ECCS performance.

Objective: To assure that the limits of Section 3.10 are not being violated.

Specification: A. Average Planar LHGR.

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

B. Local LHGR

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

C. Assembly Averaged Power-Void Relationship

Compliance with the Power-Void Relationship in Section 3.10.C will be verified at least once during a startup between 50% and 70% power, when steady state power operation is attained, and at least every 72 hours thereafter during power operation.

D. Minimum Critical Power Ratio (MCPR).

MCPR and APRM status shall be checked daily during reactor operation at $\geq$25% rated thermal power.

Basis: The LHGR shall be checked daily to determine whether fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

The Power-Void Relationship is verified between 50% and 70% power during a startup. This single verification during startup is acceptable since operating experience has shown that even under the most extreme void conditions encountered at lower power levels, the relationship is not violated. Additionally reduced power operation involves less stored heat in the core and lower decay heat rates which would add further margin to limiting peak clad temperatures in the event of a LOCA.