April 30, 1986

Docket No. 50-/324

Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 123 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit 2. The amendment consists of changes to the Technical Specifications in response to your submittal of December 20, 1985, as supplemented by submittal dated March 28, 1986.

The amendment changes the Technical Specifications (TS) by modifying the minimum critical power ration (MCPR) values and deleting references to 8x8 fuel type to support operation of Unit 2 in Fuel Cycle 7.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original signed by

Ernest D. Sylvester, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 123 to
   License No. DPR-62
2. Safety Evaluation

cc w/enclosures:
See next page

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PDR ADOCK 05000324
P PDR

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Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
License No. DPR-62

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

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated December 20, 1985, as supplemented March 28, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 123, 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

Daniel R Muller

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 30, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 123

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace lthe following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

3/4 2-1
3/4 2-2
3/4 2-3
3/4 2-4
3/4 2-5
3/4 2-6
3/4 2-7
3/4 2-8
3/4 2-9
3/4 2-10
3/4 2-11
3/4 2-12
3/4 2-13
3/4 2-14
3/4 3-42
3/4 3-82
B 3/4 2-1
B 3/4 2-3
B 3/4 2-5
5-1

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 (APLHGR's) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the following limits:

a. During two recirculation loop operation, the limits are shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5.

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

ACTION: With an APLHGR exceeding the limits of Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5, initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

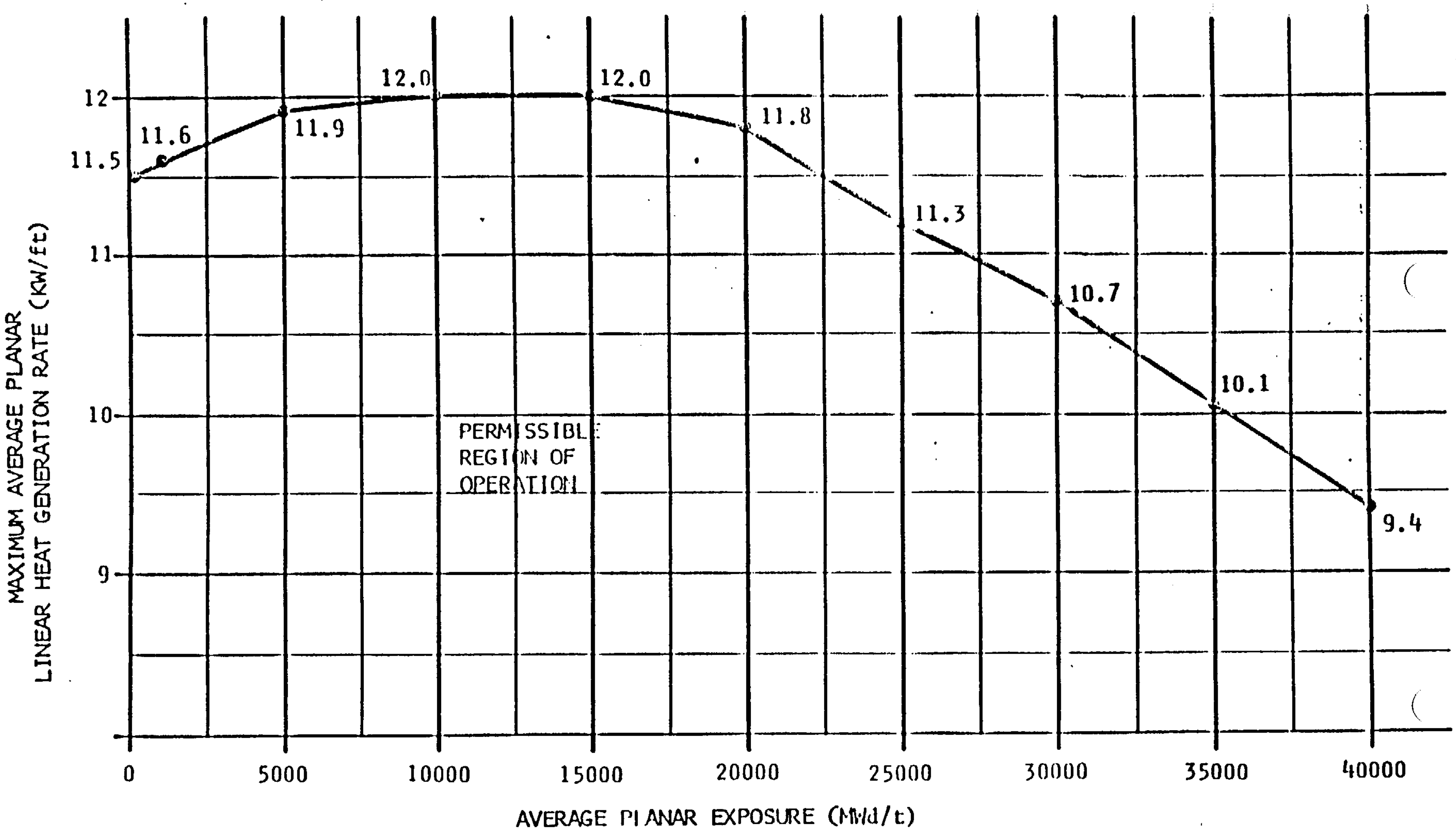
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGR's shall be verified to be equal to or less than the applicable limit determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5:

a. At least once per 24 hours,

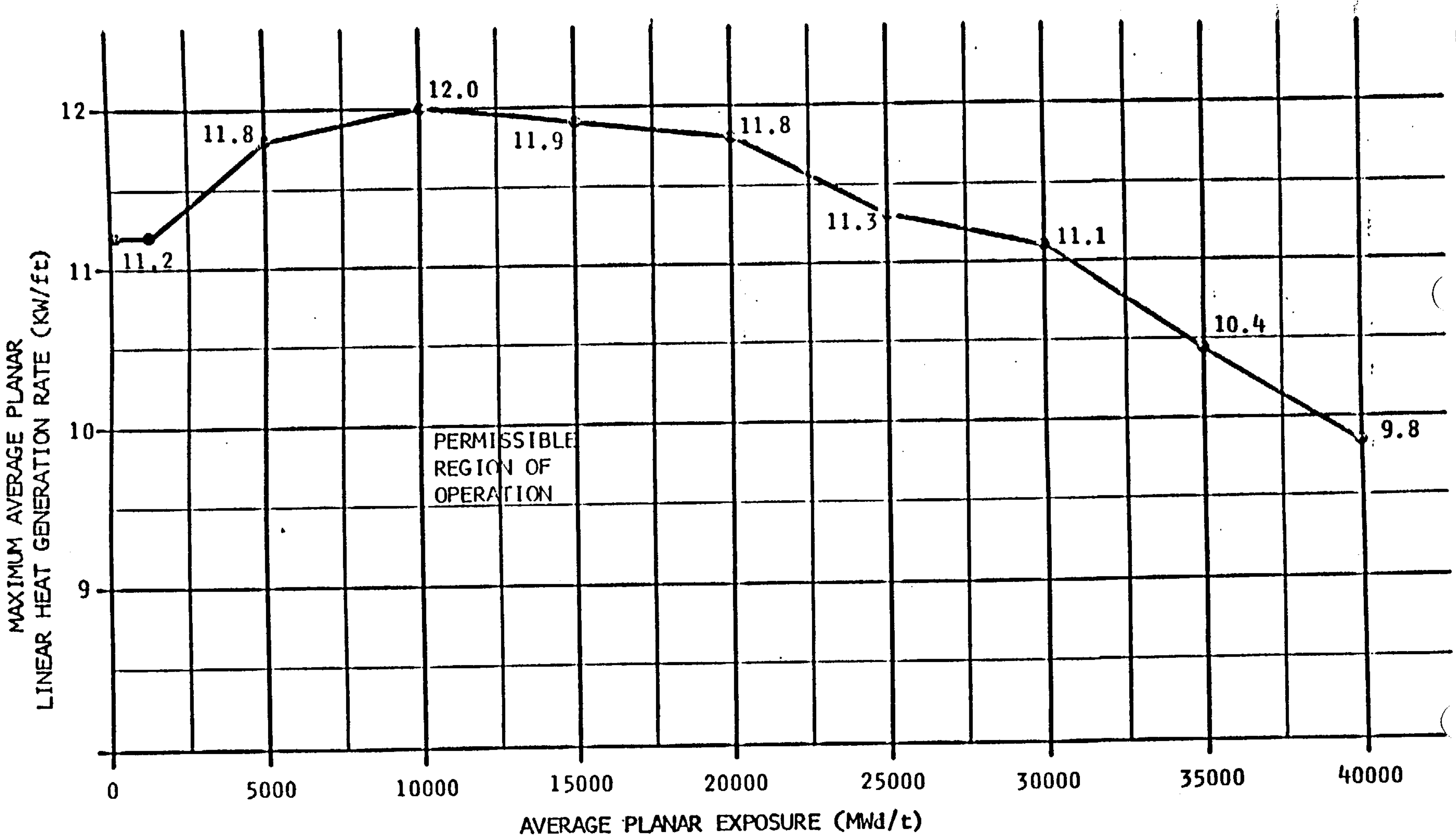
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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



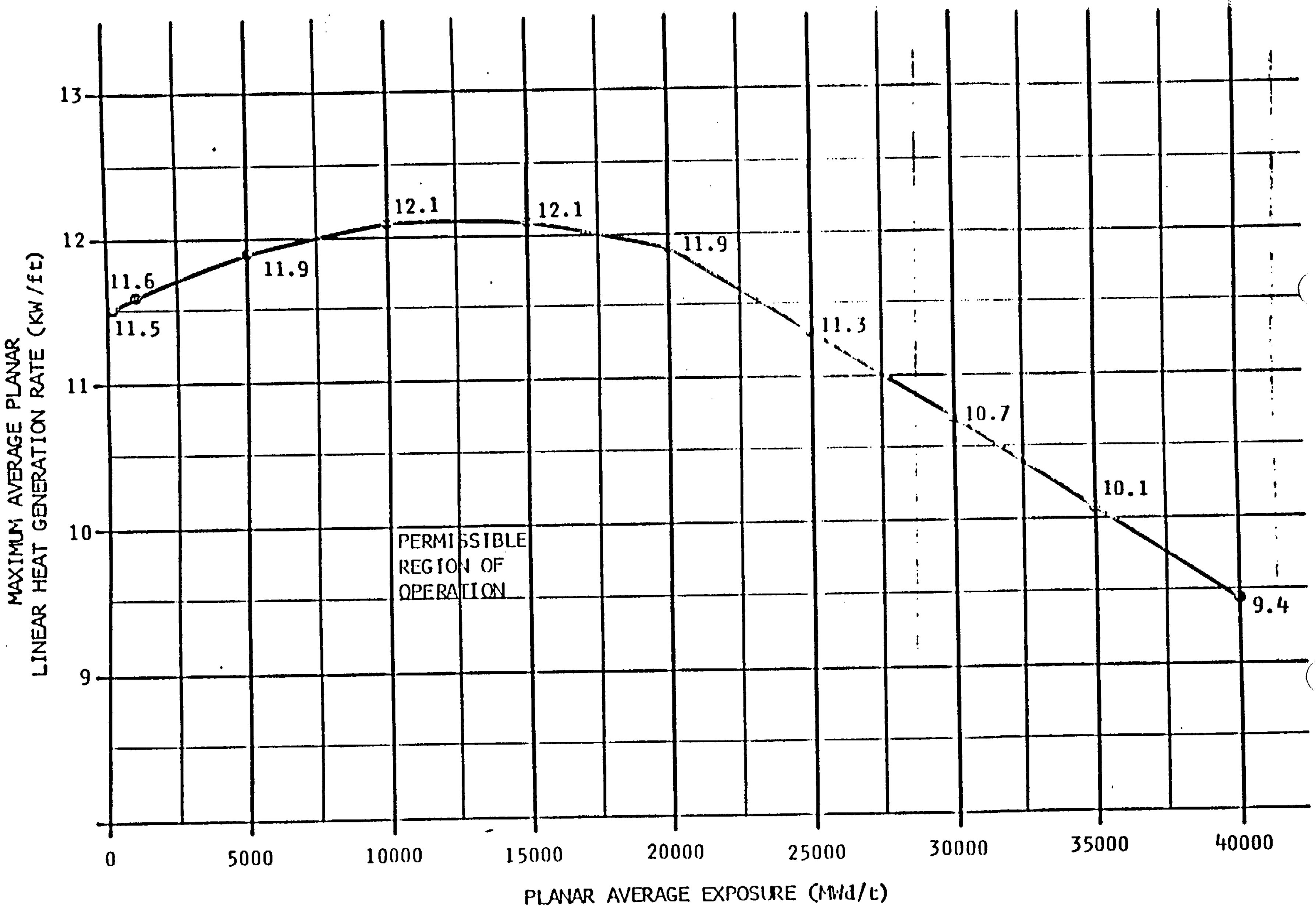
FUEL TYPE 8DRB265H (8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

Figure 3.2.1-1

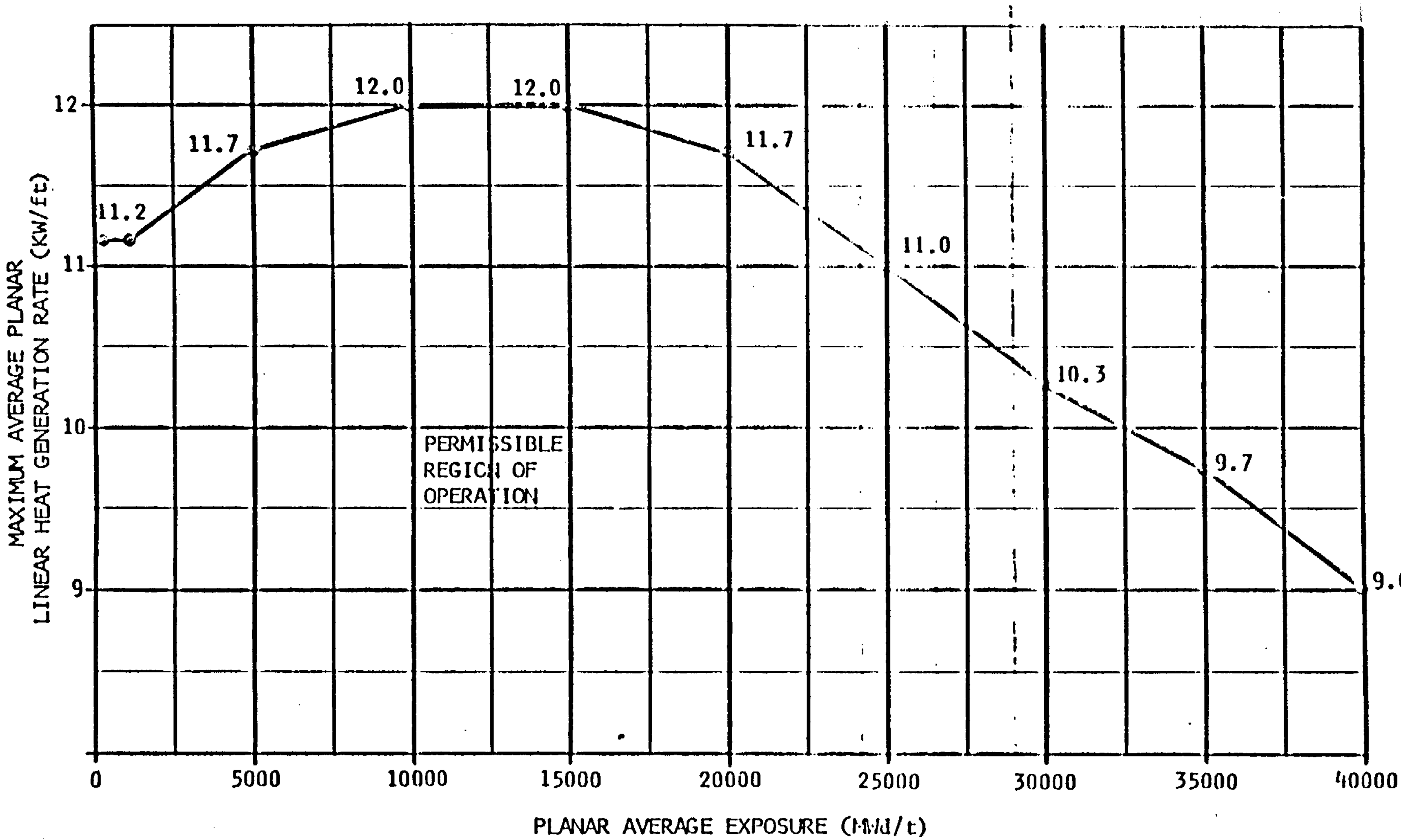


FUEL TYPE 8DRB283 (8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

Figure 3.2.1-2

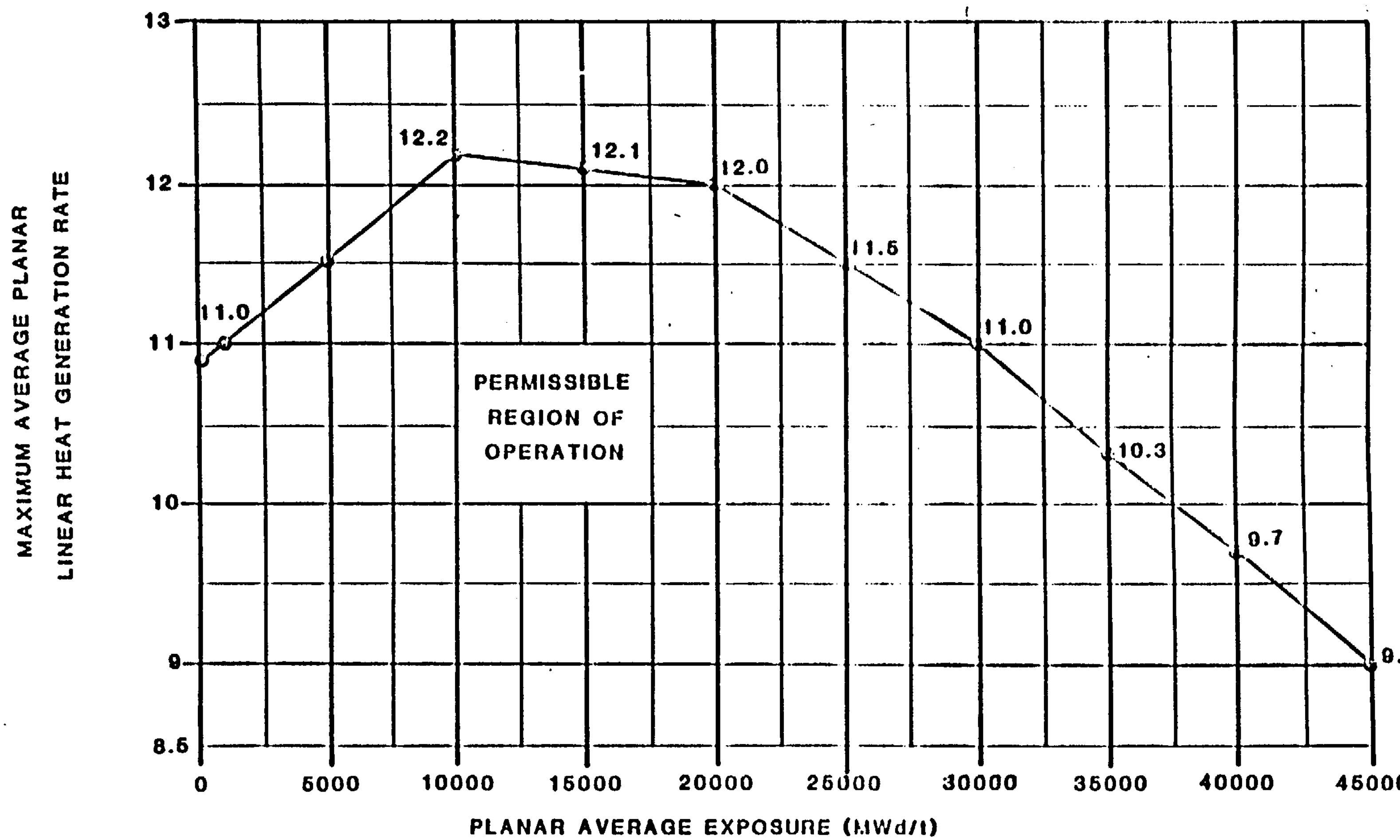


FUEL TYPE P8DRB265H (P8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE
Figure 3.2.1-3



FUEL TYPE P8DRB284H (P8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

Figure 3.2.1-4



FUEL TYPE BP8DRB209 (BP8x8R)

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

Figure 3.2.1-5

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The flow-biased APRM scram trip setpoint (S) and rod block trip set point (S_{RB}) shall be established according to the following relationship:

$$S \leq (0.66W + 54\%)\ T$$

$$S_{RB} \leq (0.66W + 42\%)\ T$$

where: S and S_{RB} are in percent of RATED THERMAL POWER.
W = Loop recirculation flow in percent of rated flow,
T = Lowest value of the ratio of design TPF divided by the MTPF obtained for any class of fuel in the core ($T \leq 1.0$), and

Design TPF for: 8 x 8R fuel = 2.39
P8 x 8R fuel = 2.39
BP8 x 8R fuel = 2.39

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

ACTION:

With S or S_{RB} exceeding the allowable value, initiate corrective action within 15 minutes and continue corrective action so that S and S_{RB} are within the required limits within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The MTPF for each class of fuel shall be determined, the value of T calculated, and the flow biased APRM trip setpoint adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3.1 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, shall be equal to or greater than the MCPR limit times the K_f shown in Figure 3.2.3-1 with the following MCPR limit adjustments:

a. Beginning-of-cycle (BOC) to end-of-cycle (EOC) minus 2000 MWD/t with ODYN OPTION A analyses in effect and the end-of-cycle recirculation pump trip system inoperable, the MCPR limits are listed below:

1. MCPR for 8 x 8R fuel = 1.31
2. MCPR for P8 x 8R fuel = 1.33
3. MCPR for BP8 x 8R fuel = 1.33

b. EOC minus 2000 MWD/t to EOC with ODYN OPTION A analyses in effect and the end-of-cycle recirculation pump trip system inoperable, the MCPR limits are listed below:

1. MCPR for 8 x 8R fuel = 1.41
2. MCPR for P8 x 8R fuel = 1.44
3. MCPR for BP8 x 8R fuel = 1.44

c. BOC to EOC minus 2000 MWD/t with ODYN OPTION B analyses in effect and the end-of-cycle recirculation pump trip system inoperable, the MCPR limits are listed below:

1. MCPR for 8 x 8R fuel = 1.29
2. MCPR for P8 x 8R fuel = 1.29
3. MCPR for BP8 x 8R fuel = 1.29

d. EOC minus 2000 MWD/t to EOC with ODYN OPTION B analyses in effect and the end-of-cycle recirculation pump trip system inoperable, the MCPR limits are listed below:

1. MCPR for 8 x 8R fuel = 1.29
2. MCPR for P8 x 8R fuel = 1.32
3. MCPR for BP8 x 8R fuel = 1.32

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

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

With MCPR, as a function of core flow, less than the applicable limit determined from Figure 3.2.3-1 initiate corrective action within 15 minutes and restore MCPR to within the applicable limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR, as a function of core flow, shall be determined to be equal to or greater than the applicable limit determined from Figure 3.2.3-1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO (ODYN OPTION B)

LIMITING CONDITION FOR OPERATION

3.2.3.2 For the OPTION B MCPR limits listed in specification 3.2.3.1 to be used, the cycle average 20% scram time (τ_{ave}) shall be less than or equal to the Option B scram time limit (τ_B), where τ_{ave} and τ_B are determined as follows:

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

i = Surveillance test number,
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test, and
τ_i = Average scram time to notch 36 for surveillance test i

$$\tau_B = \mu + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i}\right)^{1/2} (\sigma), \text{ where:}$$

i = Surveillance test number
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test
N_1 = Number of rods tested at BOC,
μ = 0.834 seconds
(mean value for statistical scram time distribution from de-energization of scram pilot valve solenoid to pickup on notch 36),
σ = 0.059 seconds
(standard deviation of the above statistical distribution).

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

LIMITING CONDITIONS FOR OPERATION (Continued)

ACTION:

Within twelve hours after determining that τ_{ave} is greater than τ_B, the operating limit MCPRs shall be either:

a. Adjusted for each fuel type such that the operating limit MCPR is the maximum of the non-pressurization transient MCPR operating limit (from Table 3.2.3.2-1) or the adjusted pressurization transient MCPR operating limits, where the adjustment is made by:

$$MCPR_{adjusted} = MCPR_{option\ B} + \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}\left(MCPR_{option\ A} - MCPR_{option\ B}\right)$$

where: τ_A = 1.05 seconds, control rod average scram insertion time limit to notch 36 per Specification 3.1.3.3,

$MCPR_{option\ A}$ = Determined from Table 3.2.3.2-1,
$MCPR_{option\ B}$ = Determined from Table 3.2.3.2-1, or,

b. The OPTION A MCPR limits listed in Specification 3.2.3.1.

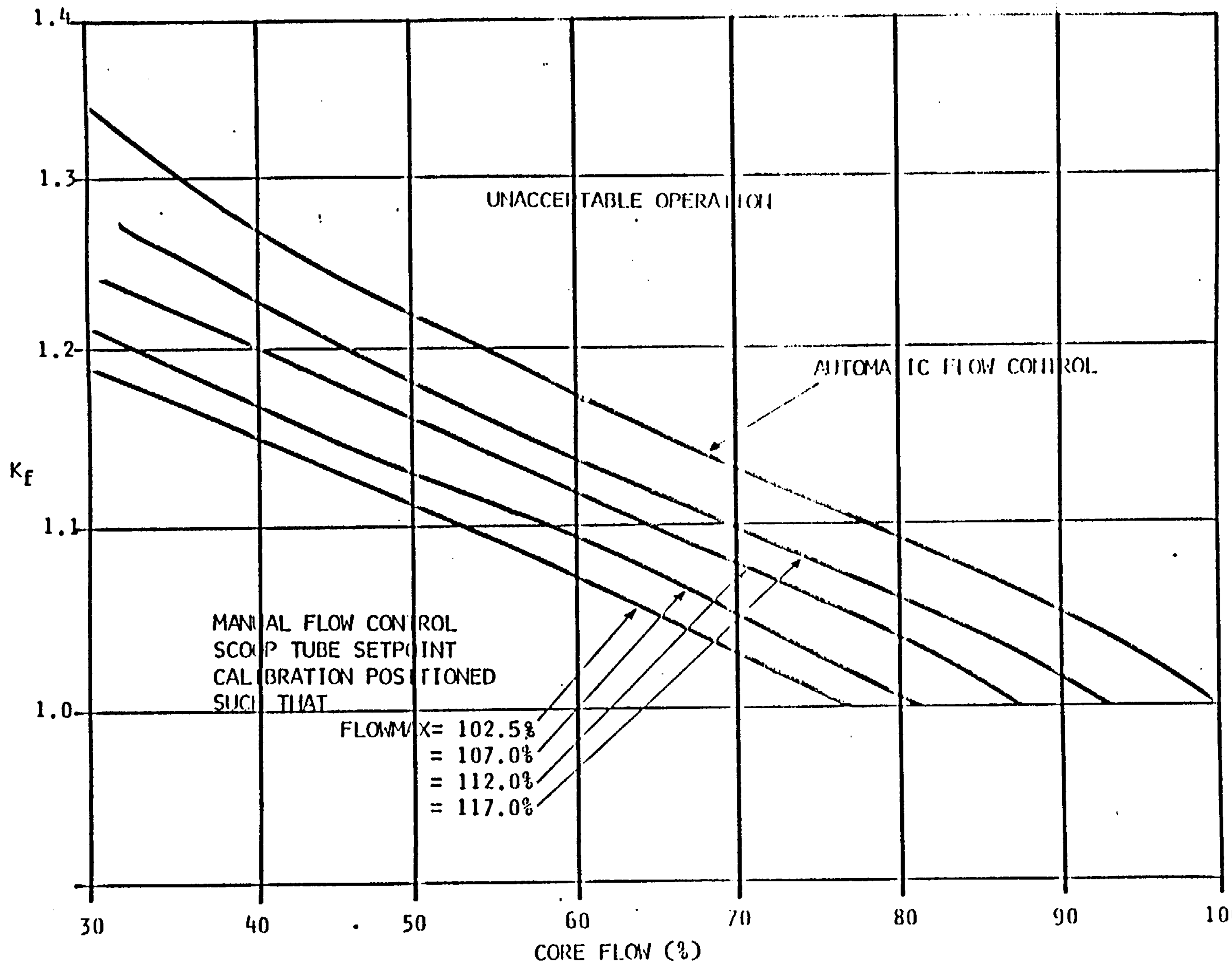
SURVEILLANCE REQUIREMENTS

4.2.3.2 The values of τ_{ave} and τ_B shall be determined and compared each time a scram time test is performed. The requirement for the frequency of scram time testing shall be identical to Specification 4.1.3.2.

TABLE 3.2.3.2-1

TRANSIENT OPERATING LIMIT MCPR VALUES

TRANSIENT	FUEL TYPE 8x8R		P8x8R		BP8 x 8R	
NONPRESSURIZATION TRANSIENTS						
BOC → EOC	1.29		1.29		1.29	
PRESSURIZATION TRANSIENTS						
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
BOC → EOC - 2000	1.31	1.17	1.33	1.17	1.33	1.17
EOC - 2000 → EOC	1.41	1.29	1.44	1.32	1.44	1.32



K_f FACTOR

FIGURE 3.2.3-1

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed 13.4 kw/ft for 8 X 8R, P8 X 8R, and BP8 x 8R fuel assemblies.

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

ACTION:

With the LHGR of any fuel rod exceeding the above limit, initiate corrective action within 15 minutes and continue corrective action so that the LHGR is within the limit within 4 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. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

TABLE 3.3.4-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. APRM (C51-APRM-CH. A,B,C,D,E,F)		
a. Upscale (Flow Biased)	$\leq (0.66W + 42\%)\frac{T^*}{MTPF}$	$\leq (0.66W + 42\%)\frac{T^*}{MTPF}$
b. Inoperative	NA	NA
c. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
d. Upscale (Fixed)	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 12% of RATED THERMAL POWER
2. ROD BLOCK MONITOR (C51-RBM-CH.A,B)		
a. Upscale	$\leq (0.66W + 39\%)\frac{T^*}{MTPF}$	$\leq (0.66W + 39\%)\frac{T^*}{MTPF}$
b. Inoperative	NA	NA
c. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
3. SOURCE RANGE MONITORS (C51-SRM-K600A,B,C,D)		
a. Detector not full in	NA	NA
b. Upscale	$\leq 1 \times 10^5$ cps	$\leq 1 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3 cps	$\geq$ 3 cps
4. INTERMEDIATE RANGE MONITORS (C51-IRM-K601A,B,C,D,E,F,G,H)		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 of full scale	$\leq$ 108/125 of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
5. SCRAM DISCHARGE VOLUME (C12-LSH-N013E)		
a. Water Level High	$\leq$ 73 gallons	$\leq$ 73 gallons

T=2.39 for 8 x 8R fuel.
T=2.39 for P8 x 8R fuel.
T=2.39 for BP8 x 8R fuel.

INSTRUMENTATION

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.6.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.6.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.*

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values Column of Table 3.3.6.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the operable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within one hour.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system operable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

* During the current cycle operation, the end-of-cycle recirculation pump trip (EOC-RPT) system will be inoperable (manually bypassed); therefore, Specification 3.3.6.2 above does not apply. The provisions of Specification 3.0.4 are not applicable.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 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 a assembly. The peak cladding temperature is calculated assuming a LHGR for the highest-powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure-dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification APHGR is this LHGR of the highest-powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5 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 Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5; (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 countercurrent flow limitation 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 Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of 2.39 for 8 x 8R, P8 x 8R, and BP8 x 8R fuel. The scram setting and rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.0 in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and peak flux indicates a TOTAL PEAKING FACTOR greater than 2.39 for 8 x 8R, P8 x 8R, and BP8 x 8R fuel. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM high flux scram curve by the reciprocal of the APRM gain change. The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and an analysis of abnormal operational transients.[1] For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient, assuming an instrument trip setting as given in Specification 2.2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

Unless otherwise stated in cycle specific reload analyses, the limiting transient which determines the required steady state MCPR limit is the turbine trip with failure of the turbine bypass. This transient yields the largest Δ MCPR. Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multichannel steady state flow distribution model as described in Section 4.4 of NEDO-20360[4] and on core parameters shown in Reference 3, response to Items 2 and 9.

MINIMUM CRITICAL POWER RATIO (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 power and flow.

The K_f factors shown in Figure 3.2.3-1 are conservative for the General Electric Plant operation with 8 x 8R fuel assembly types 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 core thermal power levels less than or equal to 25%, 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 start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level 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% 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 power or power shape, regardless of magnitude that could place operation at a thermal limit.

3.2.4 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. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

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 shall be as shown in Figure 5.1.2-1.

SITE BOUNDARY

5.1.3 The SITE BOUNDARY shall be as shown in Figure 5.1.3-1. For the purpose of effluent release calculations, the boundary for atmospheric releases is the SITE BOUNDARY and the boundary for liquid releases is the SITE BOUNDARY prior to dilution in the Atlantic Ocean.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The PRIMARY CONTAINMENT is a steel-lined, reinforced concrete 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 concrete, steel-lined pressure vessel in the shape of a torus. The primary containment has a minimum free air volume of 288,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

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

a. Maximum internal pressure 62 psig.

b. Maximum internal temperature: drywell 300°F
Suppression chamber 200°F

c. Maximum external pressure 2 psig.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 560 fuel assemblies. The 8 x 8R, P8 x 8R, BP8 x 8R fuel assemblies contain 62 fuel rods. All fuel rods shall be clad with Zircaloy 2. The nominal active fuel length of each fuel rod shall be 150 inches for 8 x 8R, P8 x 8R, and

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.123 TO FACILITY LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO 2

DOCKET NO. 50-324

1.0 INTRODUCTION

By letter dated December 20, 1985 as supplemented March 28, 1986 (Reference 1, NLS-85-415 and NLS-86-097) the Carolina Power & Light Company (CP&L, the licensee) submitted proposed changes to the Technical Specifications appended to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant (BSEP), Unit No. 2. CP&L, in the meeting with the staff on March 18, 1986, gave a technical presentation about the changes for Cycle-7. The technical information discussed in the March 18, 1986 meeting was provided formally in the March 28, 1986 submittal.

The proposed amendment would change the Technical Specifications (TSs) to permit operation of Unit 2 for Cycle 7. The changes incorporate revised minimum critical power ratio (MCPR) values and delete references to 8X8 fuel which is totally removed from the core. The Cycle-6 operating MCPR values are increased by "ADDERS" ranging from 0.04 to 0.07 ΔMCPR for Cycle-7.

The licensee has relied on the results presented in the approved GE topical report NEDE-24011, "General Electric Standard Application for Reactor Fuel", or GESTAR II (Ref. 3) for safety analyses of postulated transients and accidents, as well as for the core-related areas of fuel design, thermal-hydraulic design, nuclear design (including power distributions and reactivity analyses) and their safety analyses.

2.0 EVALUATION

2.1 Fuel System Design - Fresh Fuel Assemblies BP8DRB299

Fresh fuel assemblies (BP8DRB299), which are prepressurized 8x8 retrofit barrier fuel assemblies with an average enrichment of 2.99 w/o in U-235, will be loaded for Cycle 7 operation. Since (1) the prepressurized 8x8 retrofit barrier fuel has been previously approved (Ref. 3), and (2) the average enrichment of the fresh fuel is less than that of the approved maximum enrichment stated in Reference 3, we conclude that the fuel assemblies are acceptable for Cycle-7 operation.

2.2 Nuclear Design

The nuclear design and analysis of the proposed reload has been performed by the methods described in Reference 3. Reference 3 has been approved for use in the design and analysis of reloads in BWR reactors and its use is acceptable for this reload. We have reviewed the results of the nuclear design analysis for Brunswick Unit 2 Cycle-7 and have determined that since they are consistent with those for similar reloads and are done with acceptable methods, they are acceptable.

2.3 Thermal Hydraulic Design

The objective of the review of the thermal-hydraulic design of the core for Cycle-7 operation is to confirm that the thermal-hydraulic design has been accomplished using acceptable methods, and to assure an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated transients, and to assure that the core is not susceptible to thermal-hydraulic instability.

The review includes the following areas: (1) operating limit minimum critical power ratio (MCPR) and the related changes to the Technical Specifications, and (2) thermal-hydraulic stability. Discussion of the review concerning the thermal-hydraulic design for Cycle 7 operation follows.

2.4 Operating Limit MCPR and the Related Technical Specification Changes

The licensee performed an evaluation to establish minimum critical power ratio (MCPR) operating limits for Cycle-7. The licensee reviewed the previous reload analyses for Units 1 and 2. From that they established the limiting transients for ODYN options A and B and for different fuel exposure levels. The uncorrected ΔCPR value as calculated by GETAB for the limiting transients for BP/P8X8N fuel type was used as the base value. To this base value "ADDERS" were imposed as follows: (a) a 0.01 ΔCPR to account for the GETAB round-off process; (b) a 0.01 ΔCPR to account for mid-cycle exposure shape and scram reactivity differences between Cycles 6 and 7; (c) a 0.02 ΔCPR ADDER to provide assurance, without an adverse impact on operations, that the proposed MCPR limits bound any reasonable variation in Cycle-7 designs and potential abnormal modes of operation and (d) 0.0 to 0.03 ΔCPR to account for different fuel types. The "ADDERS" therefore varied in total from 0.04 to 0.07 depending upon the fuel type, exposure level and the type of transient. ODYN correction factors were then superimposed on the (GETAB uncorrected ΔCPR + ADDERS) values to determine the operating limit MCPR. The previous reload analyses indicate that the maximum observed cycle to cycle variation in MCPR operating limits for the limiting transients is only 0.02 ΔCPR. The proposed "ADDERS" (0.04 to 0.07 ΔCPR) are therefore conservative and are acceptable.

We find that since approved methods (Ref. 3) were used and the results show an acceptable margin of safety from conditions which could lead to fuel damage during any anticipated operational transient, that the thermal-hydraulic design of the Cycle-7 core is acceptable. The corresponding Technical Specification (3/4.2.3) changes are also acceptable since they are consistent with the Cycle-7 safety analysis.

2.5 Thermal-Hydraulic Stability

The results of thermal-hydraulic analyses show that the maximum core stability decay ratio is 0.78 for Cycle-7. We find that (1) the calculated decay ratio for Cycle-7 is less than that for similar reload cores and (2) the Technical Specifications prohibit normal operation in the natural circulation mode in which the core would be less stable. We therefore conclude that the thermal-hydraulic stability results are acceptable for Cycle-7 operation.

2.6 Transient and Accident Analyses

The Postulated Uncontrolled Rod Withdrawal Error, Fuel Misorientation Event and Rod Drop Accident have been analyzed for this cycle. The cycle specific Rod Drop Accident analysis was necessary because certain parameters (accident reactivity shape function and scram shape function in the cold startup mode) were not bounded by the generic analysis. The results of the cycle specific analysis meet our acceptance criterion (220 calories per gram peak enthalphy) for this event and are therefore acceptable.

On the basis that approved methods have been used to perform the analyses and to obtain input parameters for them and that the results of the accident analyses are acceptable for Cycle-7, we conclude that the analyses of the three cited events are acceptable. Core-wide transient analyses are discussed in Sections above.

2.7 Technical Specification Changes

Various revisions to the Technical Specifications have been proposed. The results of our review are as follows:

Section 3/4.2.3 and Table 3.2.3.2-1 of the Technical Specifications have been revised to include the proposed operating limit MCPRs for Cycle-7 operation. We find that the proposed operating limit MCPRs have been established using approved methods to avoid violation of the safety limit MCPR during any anticipated operational transient. We conclude that the Technical Specification changes related to the operating limit MCPRs are acceptable based on the discussion in Section 2.4 of this SE.

A note is added to Technical Specification 3.3.6.2 to indicate that during current Cycle operation the EOC recirculation pump trip system will be inoperable. This is acceptable since no credit is taken for this trip in the plant safety analysis.

The other changes are editorial in nature.

3.0 EVALUATION SUMMARY

From the basis of our review which is described above, we conclude that the Brunswick-2 reactor may be operated for Cycle-7 with the new fuel without undue risk to the health and safety of the public. This conclusion is based on the fact that acceptable methods and procedures were used to perform the design and analysis of the cycle and that the Technical Specifications have been correctly based on the results of that analysis.

4.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: George Thomas

Dated: April 30, 1986

5.0 References

1. Letters from CP&L to NRC, Request for License Amendment Fuel Cycle No.7-Reload Licensing, December 20, 1985, March 28, 1986 (NLS-85-415, NLS-86-097).

2. 23A1765, Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 2, Reload 5, May 1984.

3. NEDE-24011-P-A-7-US, General Electric Boiling Water Reactor Generic Reload Fuel Applications, August 1985.

4. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 6, 1981.

5. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 28, 1981.

6. L. S. Rubenstein (NRC) memorandum for T. M. Novak (NRC) on "Extension of General Electric Emergency Core Cooling System Performance Limits" dated June 25, 1981.

7. P. W. Howe (CP&L) letter to D. B. Vassallo (NRC) dated June 7, 1982.