September 22, 1984

Docket Nos. 50-325/324

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

Dear Mr. Utley:

DISTRIBUTION

Docket File	EJordan
NRC PDR	JNGrace
Local PDR	TBarnhart (8)
ORB#2 Rdg	WJones
DEisenhut	DBrinkman
SNorris	ACRS (10)
MGrotenhuis	OPA, CMiles
OELD	RDiggs
SECY	Gray File
LHarmon	Extra - 5

The Commission has issued the enclosed Amendment Nos. 75 and 101 to Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant, Units 1 and 2. The amendments consist of changes to the technical specifications in response to your submittals (NLS-84-219, NLS-84-274) of June 26, 1984.

The amendments change the Technical Specifications (TS) to permit operation of Unit 2 for Cycle 6. These changes are to incorporate revised minimum critical power ratio (MCPR) values, revised maximum average planar linear heat generation rate (MAPLHGR) values for the new BP8DRB299 fuel type, and inclusion of a footnote to TS 3.3.6.2, End-of-Cycle Recirculation Pump Trip (EOC-RPT) System Instrumentation to reflect the manual bypassing of the EOC-RPT system during Cycle 6 operation for Unit 2. In addition, TS 5.3.1 is being revised to reflect that reload fuel has a maximum enrichment of 2.99 weight percent U-235.

The amendments change the Technical Specifications, Section 5.3.2, to reflect the use of hybrid design hafnium control rod assemblies. These assemblies will be used to replace standard control rod assemblies during the current Unit 2 refueling outage and will be used as replacements during upcoming Unit 1 refuelings.

A copy of the related Safety Evaluation is also enclosed.

Sincerely,

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 75 to License No. DPR-71
2. Amendment No. 101 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures:
See next page

See note (next page) 9/10/84

DL:ORB#2	DL:ORB#2	DL:ORB#2	OELD	DL:AB-OR
SNorris:jk	MGrotenhuis	DVassallo		GLainas
09/5/84	09/6/84	09/6/84	09/ /84	09/18/84

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PDR ADOCK 05000324
P PDR

Mr. E. E. Utley
Carolina Power & Light Company
Brunswick Steam Electric Plant, Units 1 and 2

cc:

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75
License No. DPR-71

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

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated June 26, 1984, (NLS-84-219) 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-71 is hereby amended to read as follows:

8410100794 840922
PDR ADOCK 05000324
P PDR

2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 75 , 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

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: September 22, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 75

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Revise the Appendix A Technical Specifications as follows:

Remove	Insert
5-4	5-4

5.3 REACTOR CORE

FUEL ASSEMBLIES (Continued)

The initial loading shall have a maximum average enrichment of 2.35 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 2.99 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 137 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing approximately 143 inches of boron carbide, B_4C, powder or hafnium absorber rods surrounded by a cruciform-shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The nuclear boiler and reactor recirculation system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

b. For a pressure of 1250 psig, and

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 18,670 cubic feet.

5.5 METEOROLOGICAL TOWER LOCATION

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

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. 101
License No. DPR-62

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

   A. The applications for amendment by Carolina Power & Light Company (the licensee) dated June 26, 1984, (NLS-84-219 and NLS-84-274) comply 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. 101, 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

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: September 22, 1984

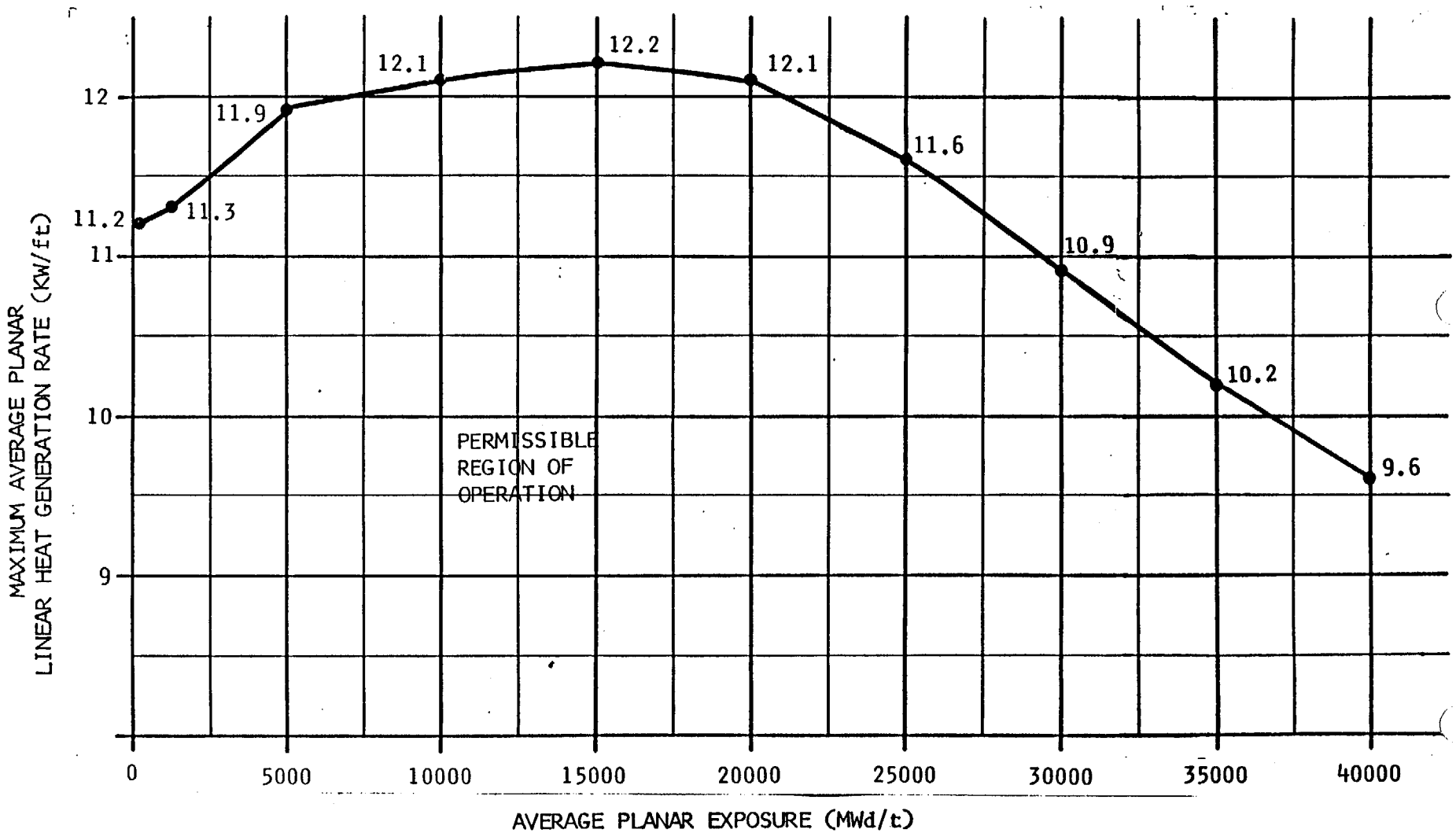
ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

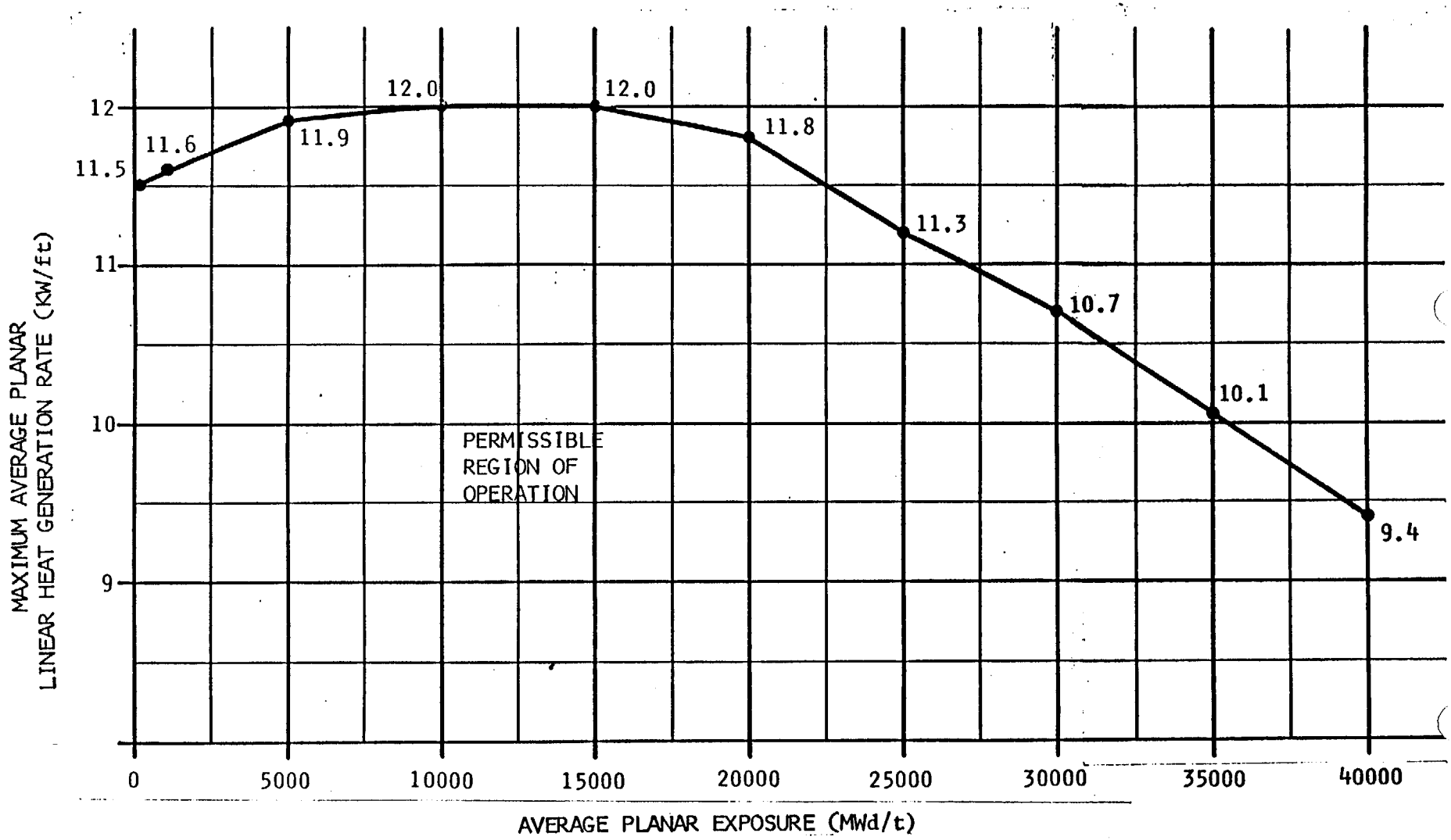
Revise the Appendix A Technical Specifications as indicated below. The changed area is indicated by vertical line.

Remove	Insert
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-13	3/4 2-13
3/4 2-15	3/4 2-15
3/4 3-42	3/4 3-42
3/4 3-82	3/4 3-82
B3/4 2-3	B3/4 2-3
5-1	5-1
5-4	5-4



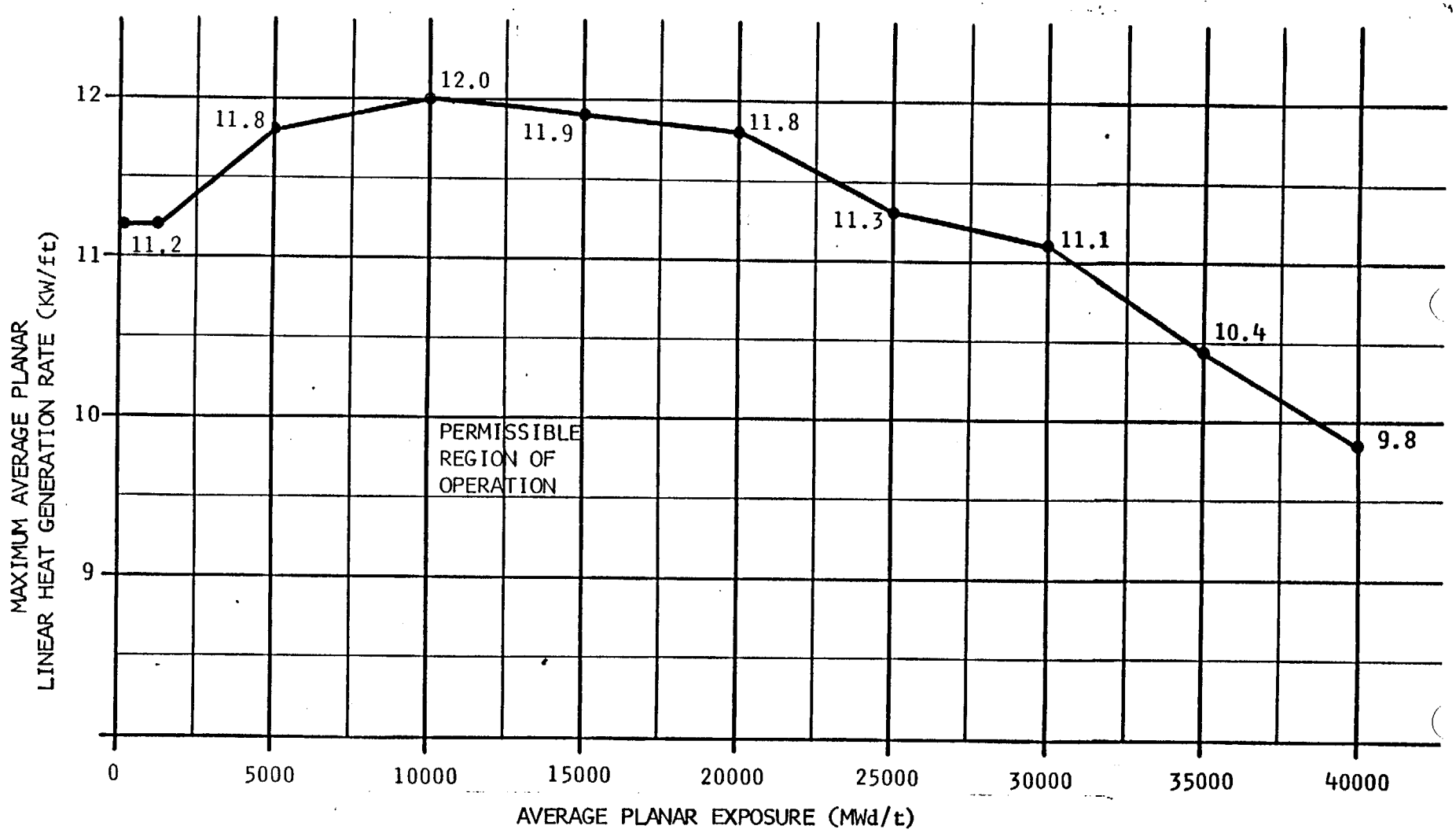
FUEL TYPE 8DB274L (8X8)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-1



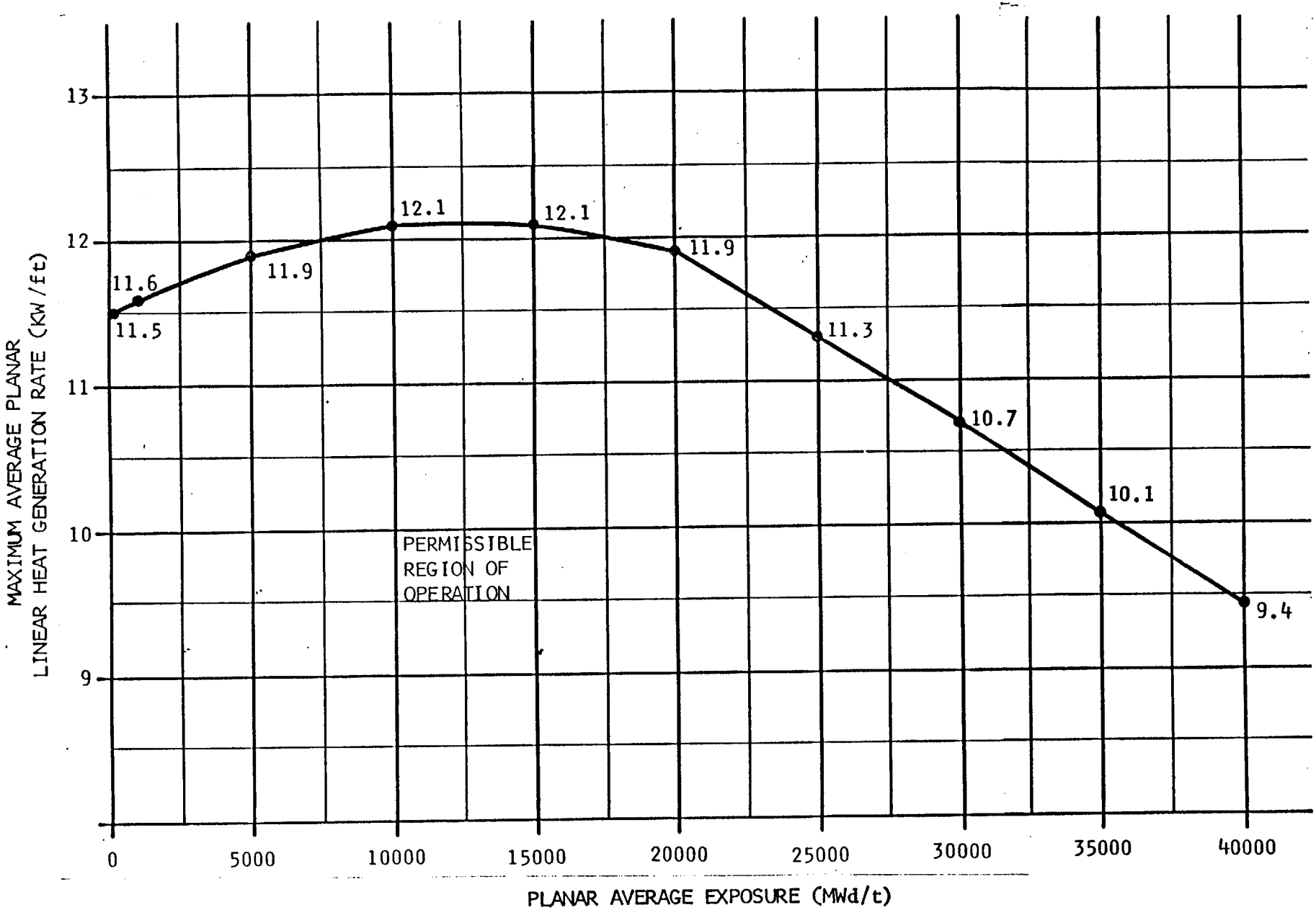
FUEL TYPE 8DRB265H (8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-2



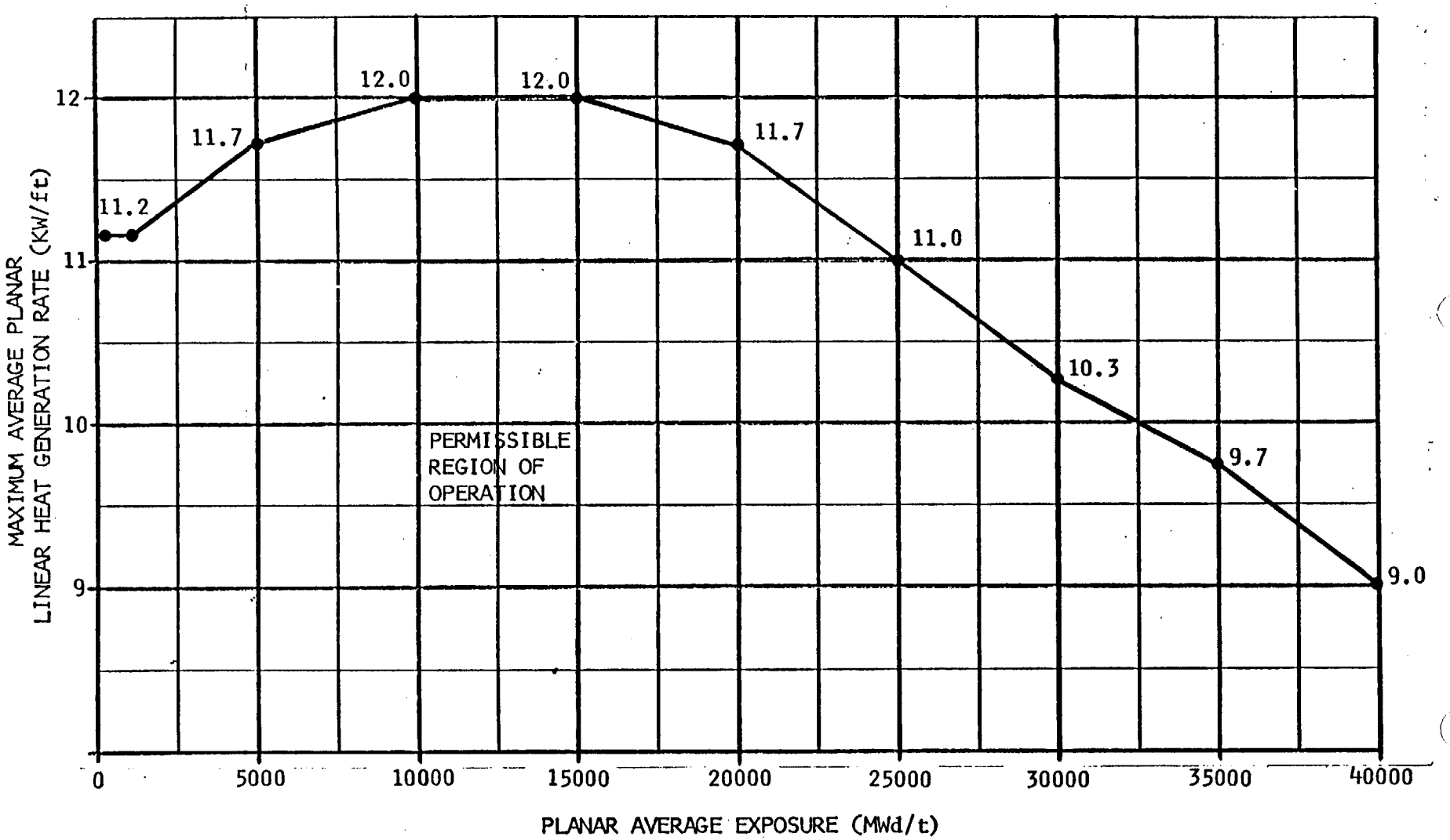
FUEL TYPE 8DRB283 (8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-3



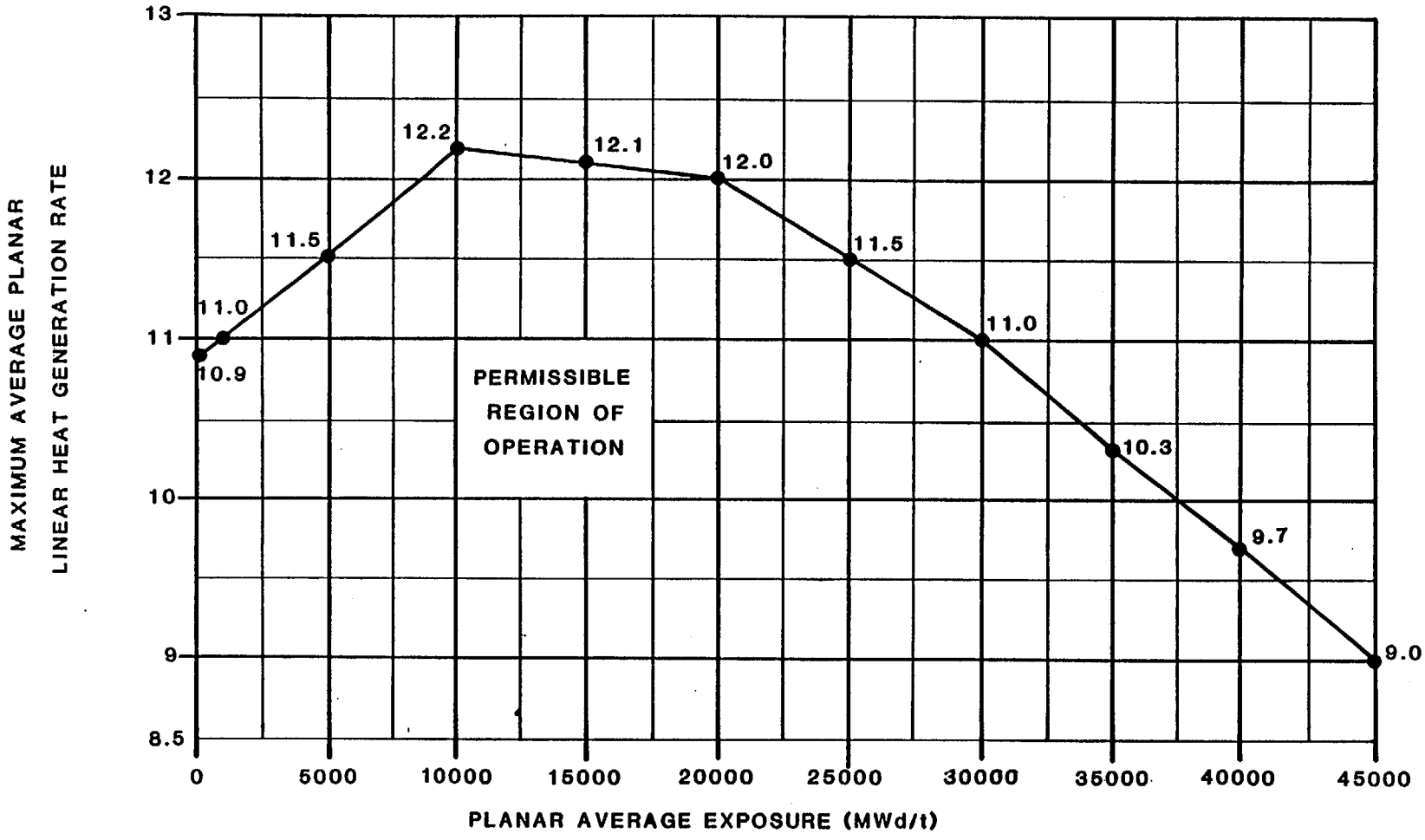
FUEL TYPE P8DRB265H (P8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE

FIGURE 3.2.1-4



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

FIGURE 3.2.1-5



FUEL TYPE BP8DRB299 (BP8x8R)

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

FIGURE 3.2.1-6

POWER DISTRIBUTION LIMITS

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 8 fuel = 2.43
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 8 fuel = 1.25
   2. MCPR for 8 x 8R fuel = 1.26
   3. MCPR for P8 x 8R fuel = 1.28
   4. MCPR for BP8 x 8R fuel = 1.28

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 8 fuel = 1.36
   2. MCPR for 8 x 8R fuel = 1.37
   3. MCPR for P8 x 8R fuel = 1.40
   4. MCPR for BP8 x 8R fuel = 1.40

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 8 fuel = 1.23
   2. MCPR for 8 x 8R fuel = 1.24
   3. MCPR for P8 x 8R fuel = 1.24
   4. MCPR for BP8 x 8R fuel = 1.24

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 8 fuel = 1.24
   2. MCPR for 8 x 8R fuel = 1.25
   3. MCPR for P8 x 8R fuel = 1.28
   4. MCPR for BP8 x 8R fuel = 1.28

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.

TABLE 3.2.3.2-1

TRANSIENT OPERATING LIMIT MCPR VALUES

TRANSIENT	FUEL TYPE 8x8		8x8R		P8x8R		BP8 x 8R	
NONPRESSURIZATION TRANSIENTS								
BOC → EOC	1.23		1.24		1.24		1.24	
TURBINE TRIP/LOAD REJECT WITHOUT BYPASS								
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
BOC → EOC - 2000	1.25	1.08	1.26	1.08	1.28	1.09	1.28	1.09
EOC - 2000 → EOC	1.36	1.24	1.37	1.25	1.40	1.28	1.40	1.28
FEEDWATER CONTROL FAILURE								
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
BOC → EOC - 2000	1.17	1.11	1.17	1.11	1.17	1.11	1.17	1.11
EOC - 2000 → EOC	1.17	1.11	1.17	1.11	1.17	1.11	1.17	1.11

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 8, 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=2.43 for 8 x 8 fuel.
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 inoperable 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 inoperable.

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 Cycle 6 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.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of 2.43 for 8 x 8 fuel and 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 2.43 for 8 x 8 fuel and 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. When added to the Safety Limit MCPR of 1.07 the required minimum operating limit MCPR of Specification 3.2.3 is obtained. 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.

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 8 fuel assemblies contain 63 fuel rods and 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 146 inches for 8 x 8 fuel assemblies and 150 inches for 8 x 8R, P8 x 8R, and

5.3 REACTOR CORE

FUEL ASSEMBLIES (Continued)

BP8 x 8R fuel assemblies. The initial core loading shall have a maximum average enrichment of 2.47 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 2.99 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 137 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing approximately 143 inches of boron carbide, B_4C, powder or hafnium absorber rods surrounded by a cruciform-shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The nuclear boiler and reactor recirculation system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 1250 psig, and

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 18,670 cubic feet.

5.5 METEOROLOGICAL TOWER LOCATION

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 75 TO FACILITY LICENSE NO. DPR-71 AND

AMENDMENT NO. 101 TO FACILITY LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 Introduction

By letters dated June 26, 2984 (Reference 1, NLS-84-219 and NLS-84-274) the Carolina Power & Light Company (the licensee) submitted proposed changes to the Technical Specifications appended to Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2.

The proposed amendments would change the Technical Specifications (TSs) to permit operation of Unit 2 for Cycle 6. The changes incorporate revised minimum critical power ratio (MCPR) values, revise maximum average planar linear heat generation rate (MAPLHGR) values for the new BP8DRB299 fuel type, and inclusion of a footnote to TS 3.3.6.2, End-of-Cycle Recirculation Pump Trip (EOC-RPT) System Instrumentation to reflect the manual bypassing of the EOC-RPT system during Cycle 6 operation for Unit 2. In addition, TS 5.3.1 is being revised to reflect that reload fuel has a maximum enrichment of 2.99 weight percent U-235. The Reload Licensing Submittal incorporates the results of analyses supporting the use of full-arc admission without Recirculation Pump Trip. This reload is based on the same General Electric generic report used in the last Unit 1 reload submittal. This report has been reviewed and approved by the staff. The fuel enrichment is identical to that for Unit 1 during the last reload for which the Commission issued Amendment No. 56 on June 28, 1983 and Amendment No. 71 on June 5, 1984.

The proposed amendments would also revise Section 5.3.2 of the Technical Specifications to reflect the use of hybrid design hafnium control rod assemblies. These assemblies will be used to replace standard control rod assemblies during the current Unit 2 refueling outage and will be used as replacements during upcoming Unit 1 refuelings. The changes made to Section 5.3.2 of the Brunswick-1 and Brunswick-2 Technical Specifications reflect the use of hybrid design hafnium control rod assemblies to replace existing control rod assemblies. The Hybrid I Control Rod (HICR) Assembly has been designed by General Electric (GE) to be used as direct replacement

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for the present control rod assemblies. The original control rods contained only boron carbide, B_4C, as the absorbing material. The new assembly design uses B_4C absorber cubes and three solid hafnium rods in the outside edge of each wing. This new design will lengthen control rod lifetime.

In the core-related areas of fuel design and safety analyses, thermal-hydraulic design and safety analysis, nuclear design including power distributions and reactivity analyses as well as safety analyses of postulated BWR accidents and transients, 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).

In addition, the licensee submitted a supplemental reload licensing document (Ref. 2) which provides results of analyses necessary to justify Cycle 6 operation but not included in GESTAR II.

2.0 Fuel System Design

2.1 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 6 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 state in Reference 3, we conclude that fuel assemblies are acceptable for Cycle 6 operation.

2.2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Limits

The licensee's submittal provided MAPLHGR limits for the 8x8, 8x8R, P8x8R and BP8x8R fuel assemblies in the Cycle 6 core. Although the methodology used is generically applicable for the MAPLHGR limit determination, the staff was concerned that the effects of enhanced fission gas release at high burnup (i.e., greater than 20 MWd/kgU) were not adequately considered in the fuel performance model. In response to this concern, GE requested (Ref. 4 and 5) that credit for approved, but unapplied, ECCS evaluation model changes and calculated peak cladding temperature margin be used to avoid MAPLHGR penalties at higher burnups. This proposal was found acceptable (Ref. 6) provided that certain plant-specific conditions were met. The licensee has stated (Ref. 7) that the GE proposal is applicable to both the Brunswick Unit 1 and Unit 2 safety analysis. We have reviewed the basis for the licensee's finding and conclude that the proposed MAPLHGR limits are appropriate for Cycle 6 operation.

2.3 Use of Hybrid I Control Rods

The licensee proposed to begin replacing the standard control rod blades with the new Hybrid I control rod (HICR) blades. These blades are designed to have the same worth and weight as the existing blades. The differences in design are in the cladding and absorber material and serve to improve blade lifetime. The use of these control rods in BWRs has been reviewed and approved by the staff (Safety Evaluation letter dated August 22, 1983) and we conclude that their use is acceptable in Brunswick-2.

The details of the design and materials will not be included in the revised Technical Specifications. Since descriptions of the standard blades exist in the FSAR and of the HICR blades in approved topical report NEDE-22290-A, and the safety design criteria which control rods must meet are contained in the FSAR and in other Technical Specifications, we conclude that this is acceptable.

3.0 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 6 and have determined that since they are consistent with those for similar reloads and are done with acceptable methods, they are acceptable.

4.0 Thermal Hydraulic Design

The objective of the review of the thermal-hydraulic design of the core for Cycle 6 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 6 operation follows:

4.1 Operating Limit MCPR and the Related Technical Specification Changes

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core will not experience boiling transition during normal operation and anticipated operational transients. As stated in Reference 3, the approved safety limit MCPR for reload cores is 1.07. A safety limit of 1.07 was used for the Cycle 6 analyses.

To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal transient, the most limiting events have been reanalyzed for this reload (Ref. 2) by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. The operating limit MCPRs for each fuel type were then established by adding the largest reduction in the minimum critical power ratio and the uncertainties associated with the calculational methods to the safety limit MCPR.

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 6 core is acceptable. The corresponding Technical Specification (3/4.2.3) changes are also acceptable since they are consistent with the Cycle 6 safety analysis.

4.2 Thermal-Hydraulic Stability

The results of thermal-hydraulic analyses (Ref. 2) show that the maximum core stability decay ratio is 0.67 for Cycle 6. We find that (1) the calculated decay ratio for Cycle 6 is less than that for similar reload cores (for example, the calculated decay ratio for the Unit 1, Cycle 4 core is 0.72) 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 6 operation.

5.0 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 (220 calories per gram peak enthalpy) meets our acceptance criterion for this event and is 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 6, we conclude that the analyses of the three cited events are acceptable. Core-wide transient analyses are discussed in Section 4.1 above.

6.0 Technical Specification Changes

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

Changes were made in Figures 3.2.1-1 through 3.2-1-6 of the Technical Specifications in order to specify the MAPLHGR limits. We conclude that these changes regarding the proposed MAPLHGR limits are acceptable based on the discussion in Section 2.1 of this SER.

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 6 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 4.1 of this SER.

In Technical Specification 3/4.2.3, the change was made in the ACTION requirements for violation of the MCPR limits. This is acceptable since the changed Technical Specification is consistent with that for the approved Unit-1 Technical Specifications.

Changes were made in Technical Specification 3/4.2.2, Table 3.3.3-2 and Bases 3/4.2.2 related to the design total peaking factors (TPF). The changes reflect the value of design TPF for Cycle 6 core and are acceptable.

Technical Specification 3/4.2.4 and Technical Specification 5.3.1 were revised to include the BP8x8R for the Cycle 6 operation and are acceptable as discussed in Section 2.2 of this report.

A note is added to Technical Specification 3.3.6.2 to indicate that during Cycle 6 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.

A change was made in Technical Specification 4.2.4 which specifies the surveillance requirements, for operation within the LHGR limit. This is acceptable since the change is to make the statement consistent with that of the standard Technical Specification and is an editorial change.

7.0 Evaluation Summary

For the basis of our review which is described above, we conclude that the Brunswick-2 reactor may be operated for Cycle 6 and that both Unit 1 and Unit 2 may be operated with the new control rods 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 the new control rods and that the Technical Specifications have been correctly based on the results of that analysis.

8.0 Environmental Considerations

The amendments involve changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet 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 amendments.

9.0 Conclusions

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: S. Sun

Dated: September 22, 1984

3.0 References

1. Letters from A. Cutter (CP&L) to D. Vassallo (NRC, Request for Revision to Technical Specifications (Fuel Cycle No. 6-Reload Licensing), June 26, 1984. (NLS-84-219, NLS-84-274).

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

3. NEDE-24011-P-A-6, General Electric Boiling Water Reactor Generic Reload Fuel Applications, April 1983.

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.