April 8, 1988

Docket No. 50-324

DISTRIBUTION
See attached sheet

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:

SUBJECT: ISSUANCE OF AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2, REGARDING FUEL CYCLE NO. 8 - RELOAD (TAC NO. 66155)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit 2 (BSEP-2). The amendment consists of changes to the Technical Specifications in response to your submittal dated September 4, 1987.

The amendment changes the Technical Specifications to incorporate the operating limits for all fuel types for Cycle 8 operation of BSEP-2. In addition, the definitions for CRITICAL POWER RATIO and PHYSICS TESTS are revised. As part of the amendment request, fuel burnups could exceed 33,000 MWD/MT. The staff has not completed its review of the environmental effects of the transportation of fuel with burnups beyond 33,000 MWD/MT. Therefore, an appropriate footnote has been added to TS Figures 3.2.1-1 through 3.2.1-5. Upon completion of the assessment, if favorable, the staff will complete the processing of the September 4, 1987 amendment request by deleting the footnote.

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,

/s/

Ernest D. Sylvester, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 149 to License No. DPR-62
2. Safety Evaluation

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P PDR

cc w/enclosures:
See next page

OFFICIAL RECORD COPY

PD21:DRPR	PD21:DRPR	PE:PD21:DRPR	PD21:DRPR
PAnderson/	ESylvester	BMozafari	EAdensam
4/7/88	4/7/88	4/7/88	4/7/88

Mr. E. E. Utley
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, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
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 September 4, 1987, 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. 149, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 8, 1988

LA:PD21:DRPR	PD:PD21:DRPR	PE:PD21:DRPR	OGC-B	D:PD21:DRPR
PAnderson	ESylvester	BMozafari		EAdensam
4/6/88	4/1/88	4/1/88	4/6/88	4/1/88

ATTACHMENT TO LICENSE AMENDMENT NO. 149

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

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

Remove Pages	Insert Pages
IV	IV
1-2	1-2
1-5	1-5
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 2-10	3/4 2-10
3/4 2-12	3/4 2-12
3/4 2-14	3/4 2-14
3/4 3-42	3/4 3-42
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL TEST (Continued)

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

CORE ALTERATION shall be the addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls in the reactor core with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative location.

CRITICAL POWER RATIO

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in an assembly which is calculated, by application of an NRC approved CPR correlation, to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be concentration of I-131, μ Ci/gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The following is defined equivalent to 1 μCi of I-131 as determined from Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites": I-132, 28 μCi; I-133, 3.7 μCi; I-134, 59 μCi; I-135, 12 μCi.

$\bar{E}$ -AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes with half lives greater than 15 minutes making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATIONAL MANUAL (ODCM) is a manual which contains the current methodology and parameters to be used to calculate offsite doses resulting from the release of radioactive gaseous and liquid effluents; the methodology to calculate gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints; and, the requirements of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION

An OPERATIONAL CONDITION shall be any one inclusive combination of mode switch position and average reactor coolant temperature as indicated in Table 1.2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and are 1) described in Section 14 of the Updated FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

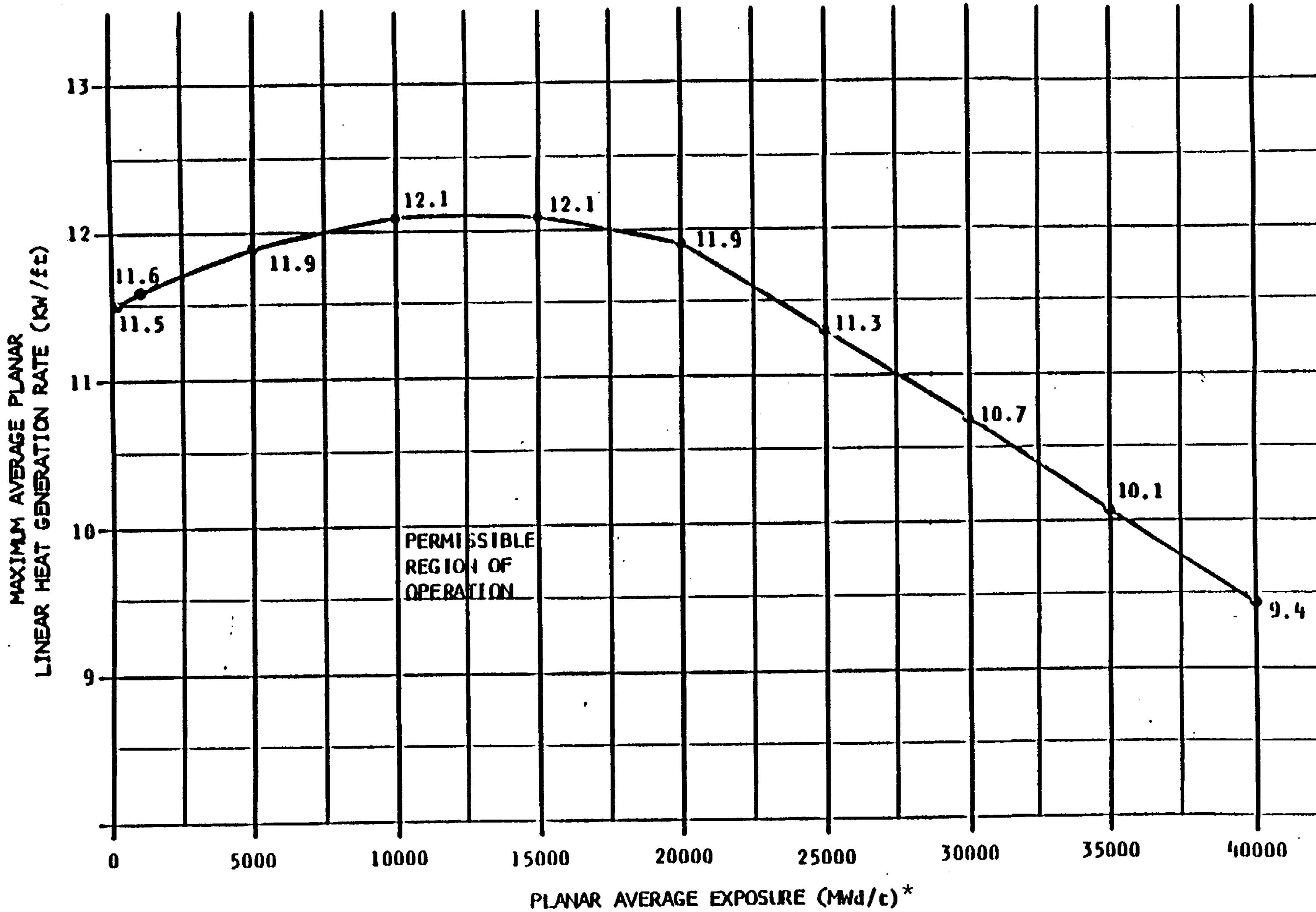
PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

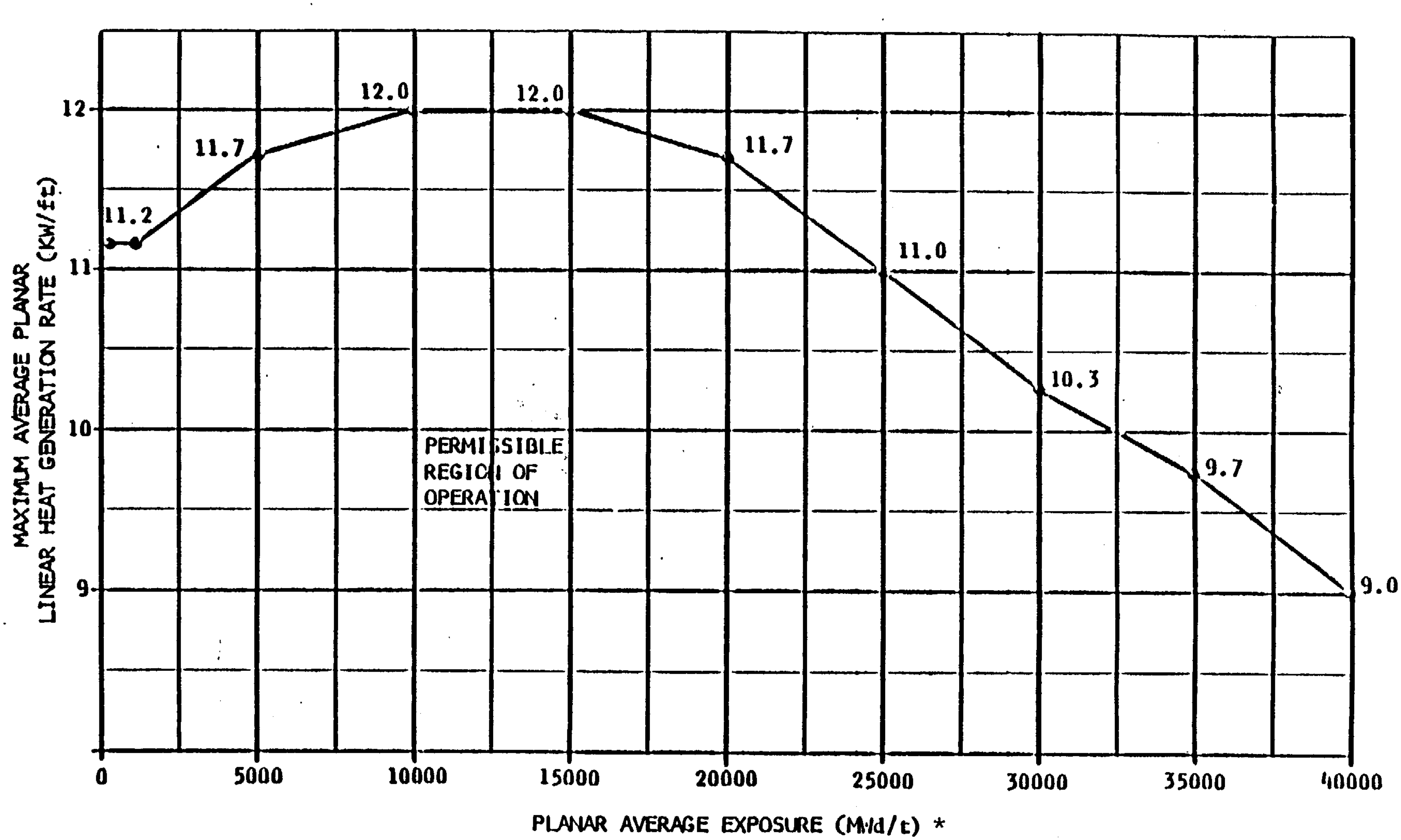
   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.1, or



*Amendment 149 authorizes operation only up to an average fuel bundle burnup of 33,000 MWD/MT

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

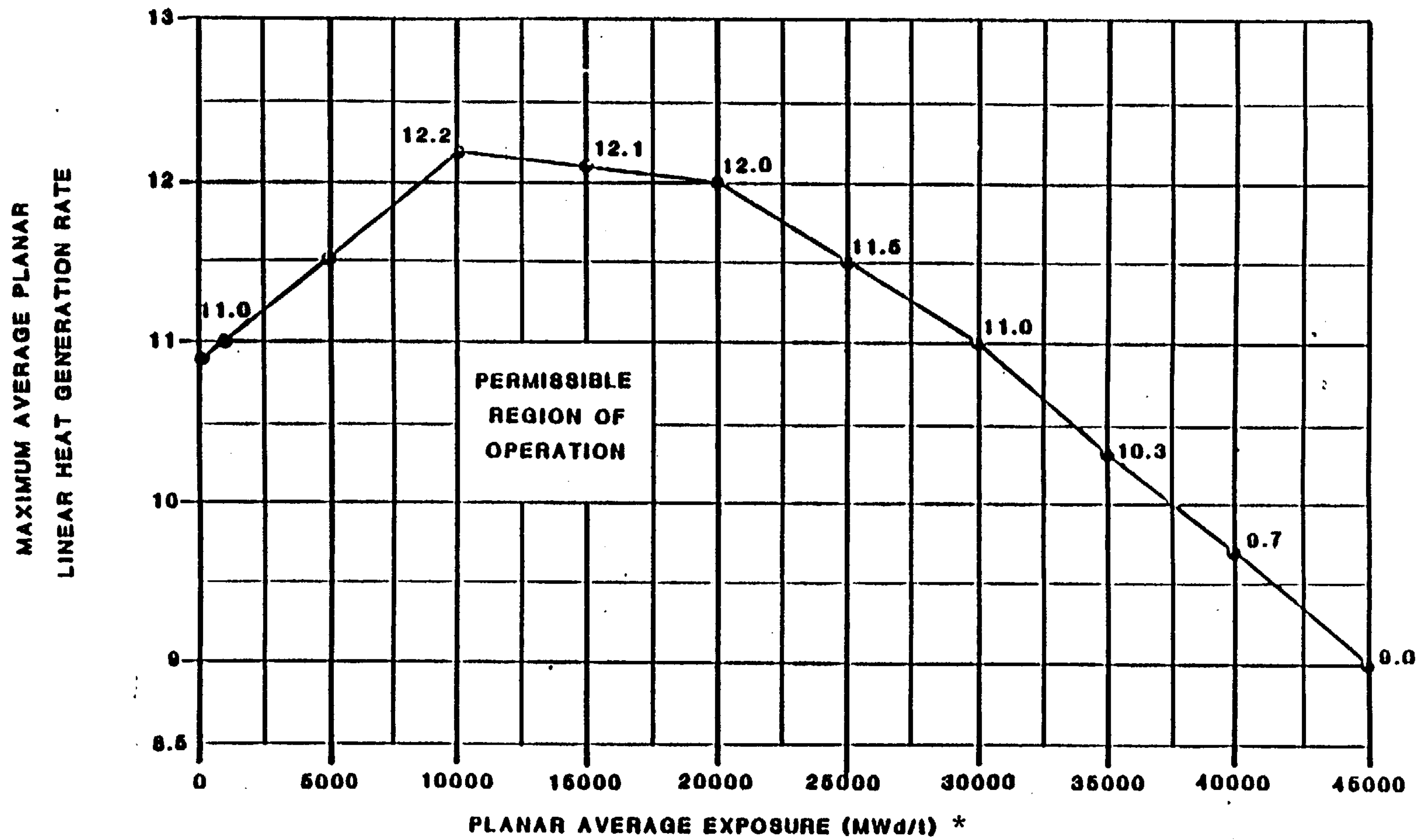
Figure 3.2.1-1



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

Figure 3.2.1-2

*Amendment 149 authorizes operation only up to an average fuel bundle burnup of 33,000 MWD/MT



FUEL TYPE BP8DRB209 (BP8x8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE

Figure 3.2.1-3

*Amendment 149 authorizes operation only up to an average fuel bundle burnup of 33,000 MWD/MT

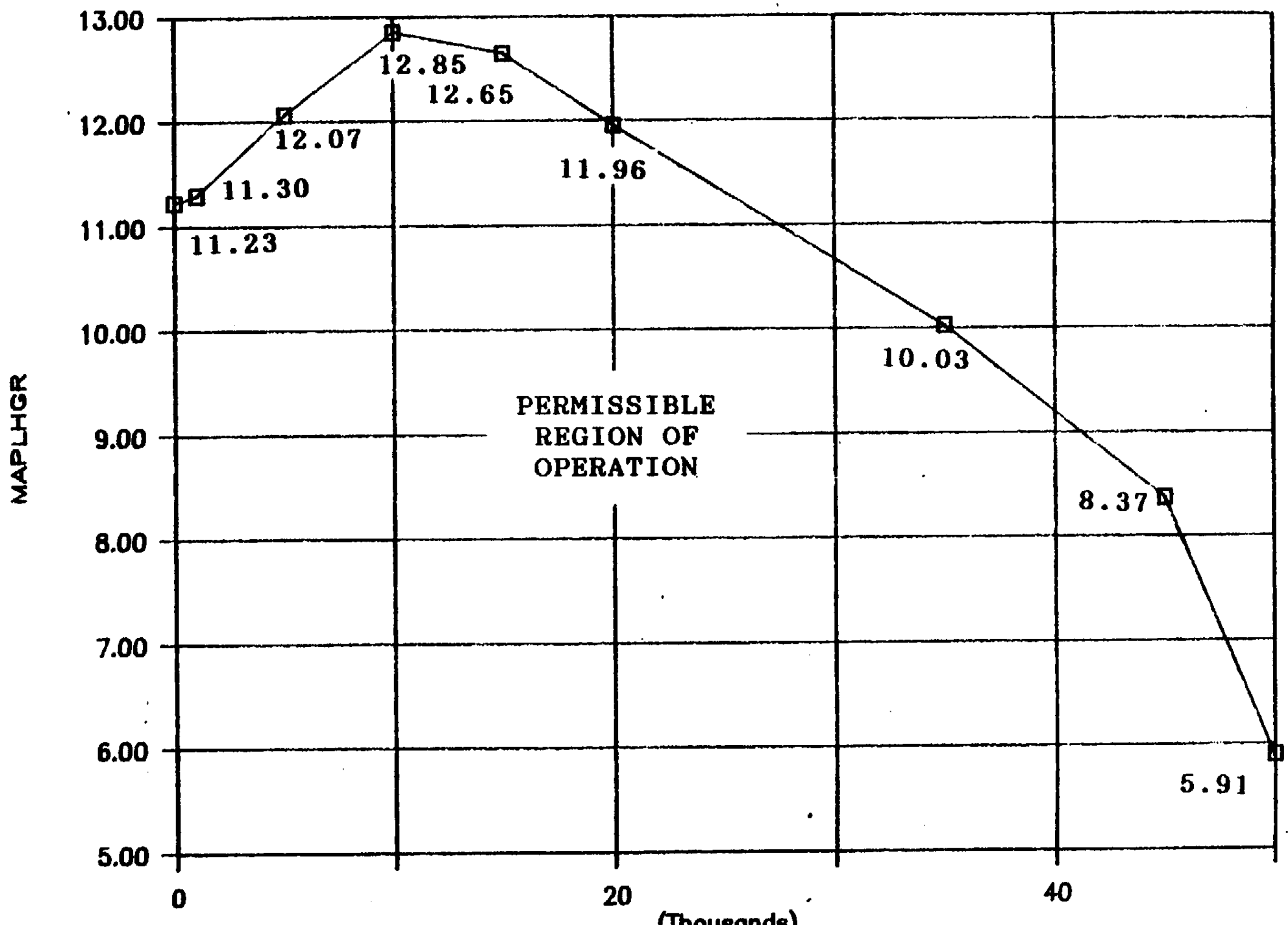


Figure 3.2.1-4

FUEL TYPE BD317A (GE8)

NOTE: This curve represents the most limiting APLHGR to be used for hand calculations. The limiting values for each lattice are in the core monitoring system.

*Amendment 149 authorizes operation only up to an average fuel bundle burnup of 33,000 MWD/MT

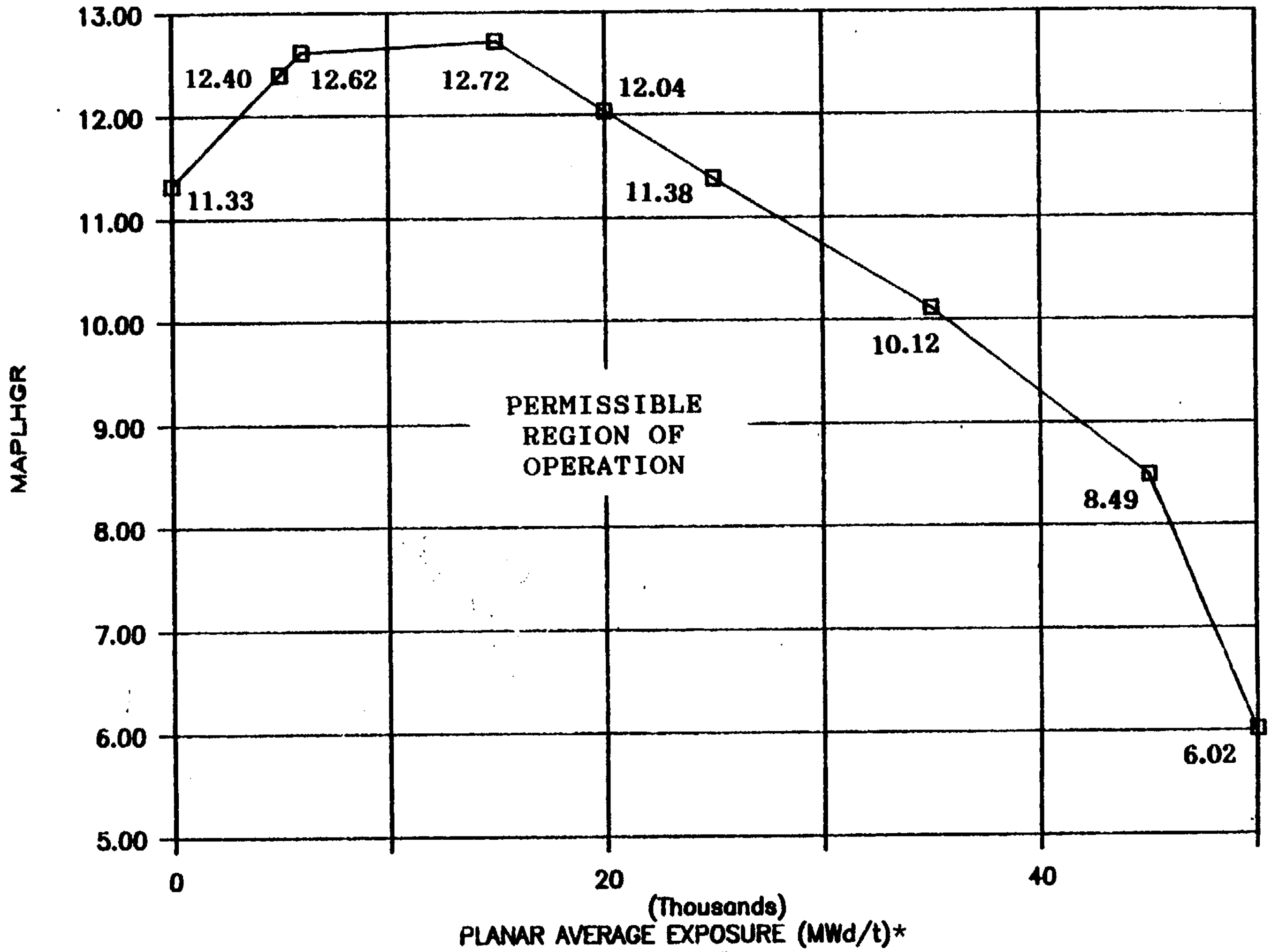


Figure 3.2.1-5

FUEL TYPE BD323A (GE8)

NOTE: This curve represents the most limiting APLHGR to be used for hand calculations. The limiting values for each lattice are in the core monitoring system.

*Amendment 149 authorizes operation only up to an average fuel bundle burnup of 33,000 MWD/MT

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 P8 X 8R fuel = 2.39
BP8 x 8R fuel = 2.39
GE8 fuel = 2.48

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.

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 P8 x 8R fuel = 1.34
2. MCPR for BP8 x 8R fuel = 1.34
3. MCPR for GE8 fuel = 1.34

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 P8 x 8R fuel = 1.35
2. MCPR for BP8 x 8R fuel = 1.35
3. MCPR for GE8 fuel = 1.35

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 P8 x 8R fuel = 1.27
2. MCPR for BP8 x 8R fuel = 1.27
3. MCPR for GE8 fuel = 1.27

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 P8 x 8R fuel = 1.31
2. MCPR for BP8 x 8R fuel = 1.31
3. MCPR for GE8 fuel = 1.31

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

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% (notch 36) 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 = 0.813 seconds
(mean value for statistical scram time distribution from de-energization of scram pilot valve solenoid to pickup on notch 36),
σ = 0.018 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.

TABLE 3.2.3.2-1

TRANSIENT OPERATING LIMIT MCPR VALUES

TRANSIENT	FUEL TYPE P8x8R		BP8x8R		GE8	
NONPRESSURIZATION TRANSIENTS						
BOC → EOC	1.27		1.27		1.27	
PRESSURIZATION TRANSIENTS						
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
BOC → EOC - 2000	1.34	1.27	1.34	1.27	1.34	1.27
EOC - 2000 → EOC	1.35	1.31	1.35	1.31	1.35	1.31

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 P8x8R and BP8x8R fuel assemblies and 14.4 kw/ft for GE8 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%) T[(a)]	$\leq$ (0.66W + 42%) T[(a)]
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%) T[(a)]	$\leq$ (0.66W + 39%) T[(a)]
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

(a) T as defined in Specification 3.2.2.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR BRUNSWICK - UNIT 2

Plant Parameters;

Core Thermal Power	2531 Mwt which corresponds to 105% of rated steam flow
Vessel Steam Output	10.96×10^6 Lbm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area for Large Breaks	
a. Discharge	2.4 ft^2 (DBA); 1.9 ft^2 (80% DBA)
b. Suction	4.2 ft^2
Number of Drilled Bundles	520

Fuel Parameters:

FUEL TYPES	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER** RATIO
Reload Core	BP/P8x8R	13.4	1.4	1.20
	GE8x8EB	14.4	1.4	1.20

A more detailed list of input to each model and its source is presented in Section II of Reference 1.

* This power level meets the Appendix K requirement of 102%.

** To account for the 2% uncertainty in bundle power required by Appendix K, the SCAT calculation is performed with an MCPR of 1.18 (i.e., 1.2 divided by 1.02) for a bundle with an initial MCPR of 1.20.

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 P8x8R and BP8x8R fuel and 2.48 for GE8 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 P8x8R and BP8x8R fuel and 2.48 for GE8 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY, et al.

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

DOCKET NO. 50-324

1.0 INTRODUCTION

By letter dated September 4, 1987, the Carolina Power & Light Company submitted a request for changes to the Brunswick Steam Electric Plant, Unit 2, (BSEP-2) Technical Specifications (TS) to incorporate operating limits using General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies.

The amendment relates to the inclusion of new and/or revised Minimum Critical Power Ratio (MCPR) limits, Average Power Range Monitor (APRM) setpoints, Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits, and Linear Heat Generation Rate (LHGR) limits for all of the fuel using Cycle 8 core and transient parameters. The new fuel is the extended burnup type, which has been used in several recent GE reloads.

This evaluation does not address the acceptability of fuel with a burnup rate beyond 33,000 MWD/MT with respect to the environmental effects of transportation. Specifically, the environmental effects of the transportation of fuel with a higher burnup rate is still being reviewed. Therefore, a footnote has been added to TS figures 3.2.1-1 through 3.2.1-5. Upon completion of its assessment, if favorable, the staff will complete the processing of the September 4, 1987 amendment requested by deleting the footnotes.

2.0 EVALUATION

2.1 Reload Description

The BSEP-2, Cycle 8 reload will retain 44 P8x8R and 332 BP8x8R GE fuel assemblies from the previous cycle and add 184 new GE8x8EB fuel assemblies. The reload is based on a previous cycle core nominal average exposure of 20,449 megawatt days per metric ton (MWD/MT) and Cycle 8 end of cycle (EOC) exposure of 20,814 MWD/MT. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery. This loading is acceptable to the NRC staff.

2.2 Fuel Design

The new fuel for Cycle 8 is the GE extended burnup fuel GE8x8EB. The fuel designations are BD317A and BD232A. This fuel type has been approved in the

NRC Safety Evaluation Report for Amendment 10 to GESTAR II. The specific descriptions of this fuel have been submitted in Amendment 18 to GESTAR II. However, since this amendment has not as yet been accepted, the fuel description has also been presented for Brunswick 2, Cycle 8, in a letter from S. R. Zimmerman (CP&L) to NRC dated October 2, 1987. This fuel description is acceptable.

The proposed Linear Heat Generation Rate (LHGR) for the GE8x8EB fuel is 14.4 kilowatts per foot (kw/ft) as compared to 13.4 kw/ft for the other GE fuel. This LHGR has been reviewed and accepted for this fuel in the GE extended burnup fuel review. This LHGR is, therefore, acceptable as the new fuel in Cycle 8.

2.3 Nuclear Design

The nuclear design analyses for Cycle 8 have been performed by GE with the approved methodology described in GESTAR II. The results of these analyses are given in the GE reload report in standard GESTAR II format. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 1.2% delta k at both beginning of cycle (BOC) and at the exposure of minimum shutdown margin, thus fully meeting the required 0.38% delta k. Since these and other Cycle 8 nuclear design parameters have been obtained using previously approved methods, and fall within expected ranges, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design analyses for Cycle 8 have been performed by GE with the approved methodology described in GESTAR II and the results are given in the GE reload report. The parameters used for the analyses are those approved for the Brunswick class BWR 4. The GEMINI system of methods was used for relevant transient analyses. The revised constants mu and sigma, which are a part of the TS changes for Cycle 8 (Specification 3.2.3.2), are used to calculate the ODYN Option B scram time limit, conforming to the approved GEMINI/ODYN analysis methods. These revised constants are appropriate for 20% scram insertion time requirements where control rod notch position 36 corresponds to the 20% scram time position.

The Operating Limit MCPR (OLMCPR) values are determined by the limiting transients, which are usually Rod Withdrawal Error (RWE), Feedwater Controller Failure (FWCF), and Load Rejection Without Bypass (LRWBP). The analyses of these events for Cycle 8, using the standard, approved ODYN Option A and B approach for pressurization transients, provide new Cycle 8 Technical Specification values of OLMCPR in the standard operating region.

For Cycle 8, the licensee follows standard practice by having exposure dependent OLMCPR values. Two exposure regions from BOC to EOC were analyzed: (1) BOC to EOC - 2 GWD/ST, and (2) EOC - 2 GWD/ST to EOC. For all standard operating conditions, LRWBP is controlling at both Option A and B limits. These OLMCPR results are reflected in TS changes, which also include an adder of 0.02 to support extended periods of operation during operational conditions, such as a main steam line isolation valve out-of-service event or a feedwater heater out-of-service event.

The licensee has performed analyses which show that an adder of 0.02 to the proposed MCPR limits conservatively bounds these abnormal modes of operation. Approved methods were used to analyze these events, as well as others which could be limiting, and the analyses and results are acceptable and fall within expected ranges.

The BSEP-2 TS contain requirements approved by NRC staff for the detection and suppression of core thermal-hydraulic instability for two or one recirculation loop operation. These specifications reflect the conclusions of the NRC Generic Letters 86-02 and 86-09, which were based on extensive stability reviews and the recommendations of the GE report SIL-380. Thus, cycle specific stability calculations are not required for Cycle 8 operation.

2.5 Transient and Accident Analyses

The transient and accident analysis methodologies used for Cycle 8 are described, and NRC approval indicated, in GESTAR II. The GEMINI system of methods option was used for transient analyses. The limiting MCPR events for BSEP-2, Cycle 8, are indicated in Section 2.4. The core wide transient analysis methodologies and results are acceptable and fall within expected ranges.

The RWE was analyzed on a plant and cycle specific basis (as opposed to the statistical approach) and a rod block setpoint of 107 was selected to provide an OLMCPR of 1.25 for all fuel types. The fuel misorientation event was analyzed with standard methods for the D lattice fuel, giving a nonlimiting MCPR of 1.20. The results of the cycle specific control rod drop accident from both cold conditions and hot standby conditions meet the NRC acceptance criterion (220 calories per gram peak enthalpy) for this event. The local transient event analyses have been analyzed with approved methods and acceptable input assumptions and result in acceptable consequences for Cycle 8.

The limiting pressurization event, the main steam isolation valve closure with flux scram, analyzed with standard GESTAR II methods, gave results for peak steam dome and vessel pressures well below required limits. These are acceptable methodologies and results.

Loss-of-coolant-accident analyses, using approved methodologies (SAFE/REFLOOD) and parameters, were performed to provide MAPLHGR values for the new reload fuel assemblies (GE8x8EB). The results are within the limits of 10 CFR 50.46 and are, therefore, acceptable.

The staff has reviewed the reports submitted for the Cycle 8 operation of BSEP-2. Based on this review, the staff concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design, and transient and accident analyses are acceptable. There are also minor administrative changes to the index, pagination, the definitions of CRITICAL POWER RATIO and PHYSICAL TESTS, associated Bases, and references. These are all acceptable. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

3.0 ENVIRONMENTAL CONSIDERATIONS

This 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. 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 off site; and that there should be no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this 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.

4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the FEDERAL REGISTER (53 FR 2310) on January 27, 1988, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

The staff has 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.

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AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

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