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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. DPR-71

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

   A. The applications for amendment by Carolina Power & Light Company (the licensee) dated May 23, May 30, as supplemented June 4, and June 25, 1980 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-71 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 29, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8007140 614

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 1, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages. The changed area of the revised page is reflected by a marginal line.

Remove	Insert
III/IV	III/IV
V/VI	V/VI
3/4 2-1/2	3/4 2-1/2
3/4 2-5/6	3/4 2-5/6
3/4 2-7/8	3/4 2-7/8
3/4 2-9/10	3/4 2-9/10
-	3/4 2-11
3/4 3-41/42	3/4 3-41/42
B3/4 2-1/2	B3/4 2-1/2
B3/4 2-3/4	B3/4 2-3/4
3/4 4-4	3/4 4-4

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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 limits shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 or 2.3.1-6.

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 or 3.2.1-6, 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.

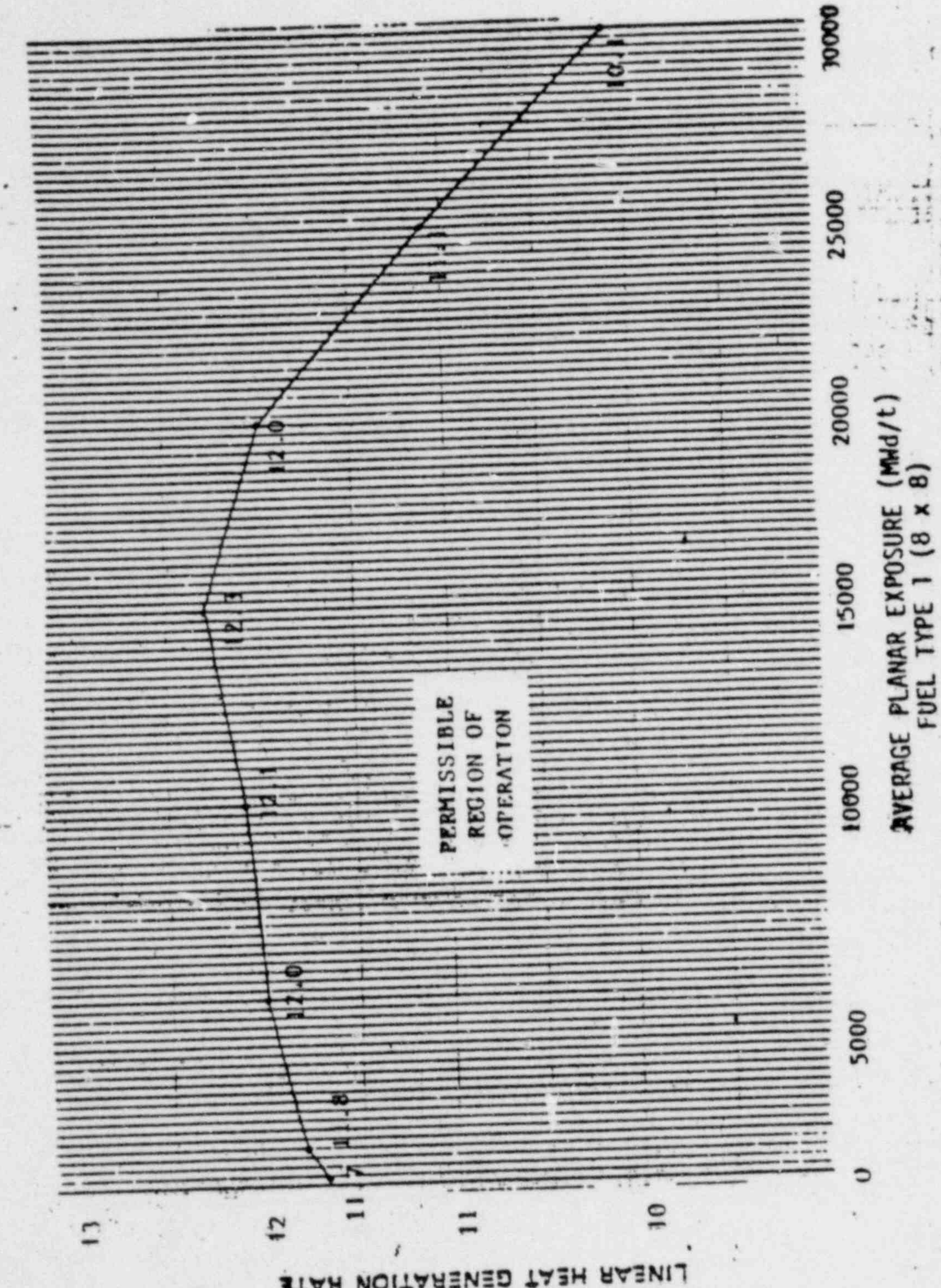
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGR's shall be verified to be equal to or less than the applicable limit determined from Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 or 3.2.1-6:

a. At least once per 24 hours,

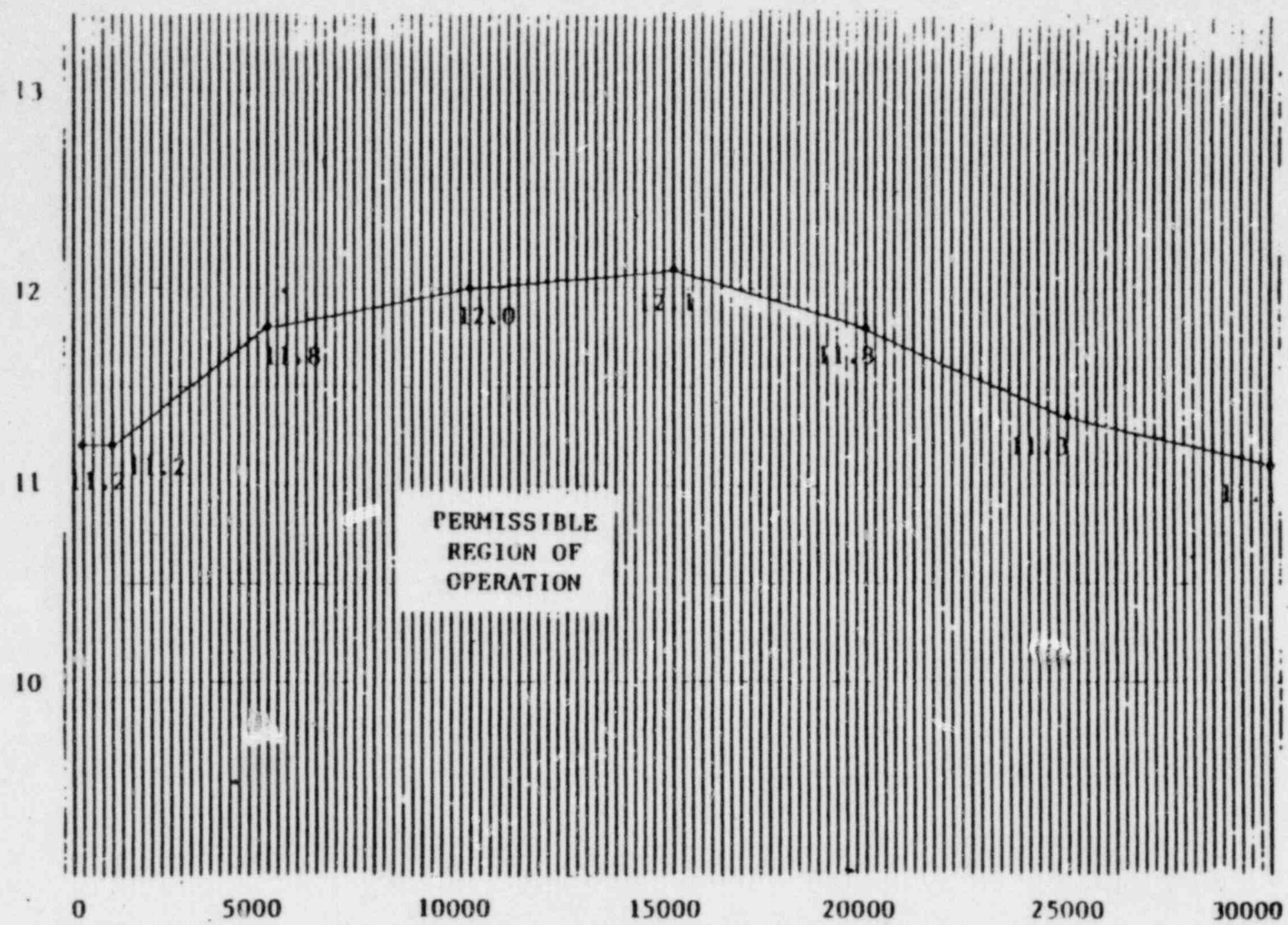
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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



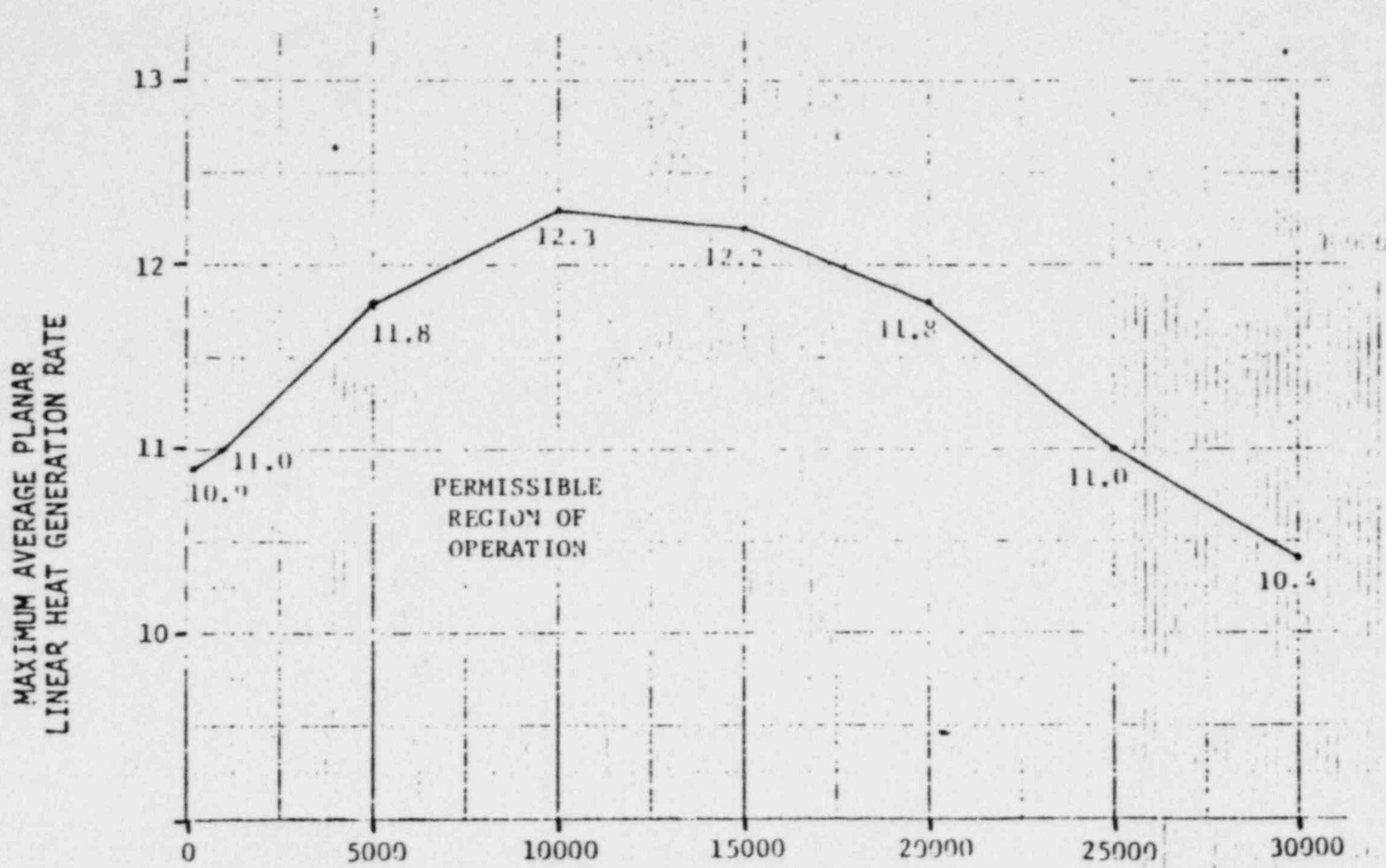
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

Figure 3.2.1-1



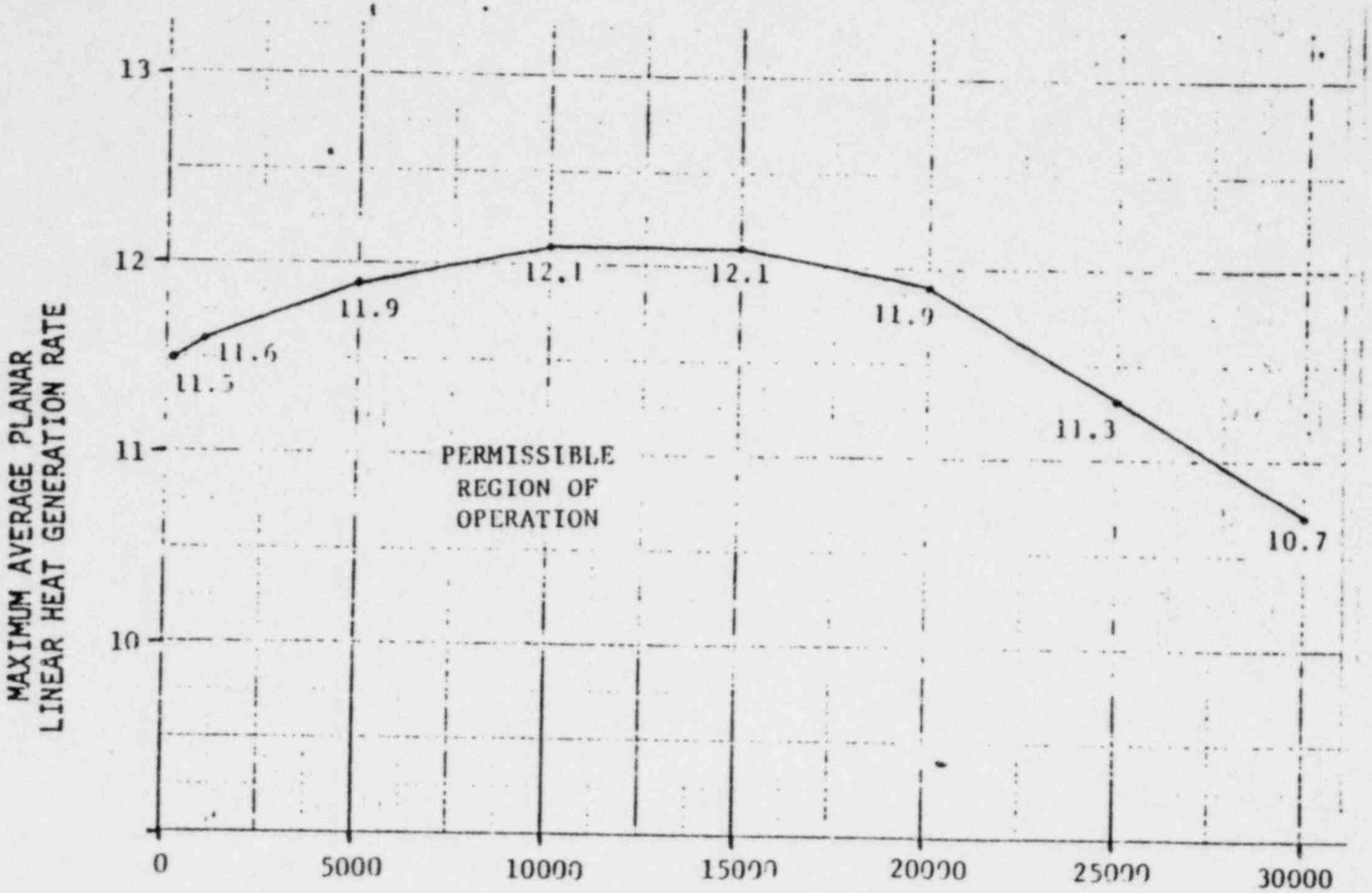
PLANAR AVERAGE EXPOSURE (MWd/t)
FUEL TYPE 8DRB283 (8x8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-4



PLANAR AVERAGE EXPOSURE (MWd/t)
FUEL TYPE P8DRB285 (P8x8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-5



PLANAR AVERAGE EXPOSURE (MWd/t)
FUEL TYPE P8DRB285 (P8x8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-6

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 setpoint (S_{RB}) shall be established according to the following relationships:

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

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

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow in percentor 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 = 2.45.
8 x 8R fuel = 2.48.
P8 x 8R fuel = 2.48.

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 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 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, shall be equal to or greater than MCPR x the K_f shown in Figure 3.2.3-1 where MCPR values are:

	BOC3* to EOC3** -2000 MWD/t	EOC3-2000 MWD/t to EOC3
8x8 fuel	1.24	1.30
8x8R fuel	1.24	1.30
P8x8R fuel	1.30	1.32

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% RATED THERMAL POWER

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 continue corrective action so that MCPR is equal to or greater than 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

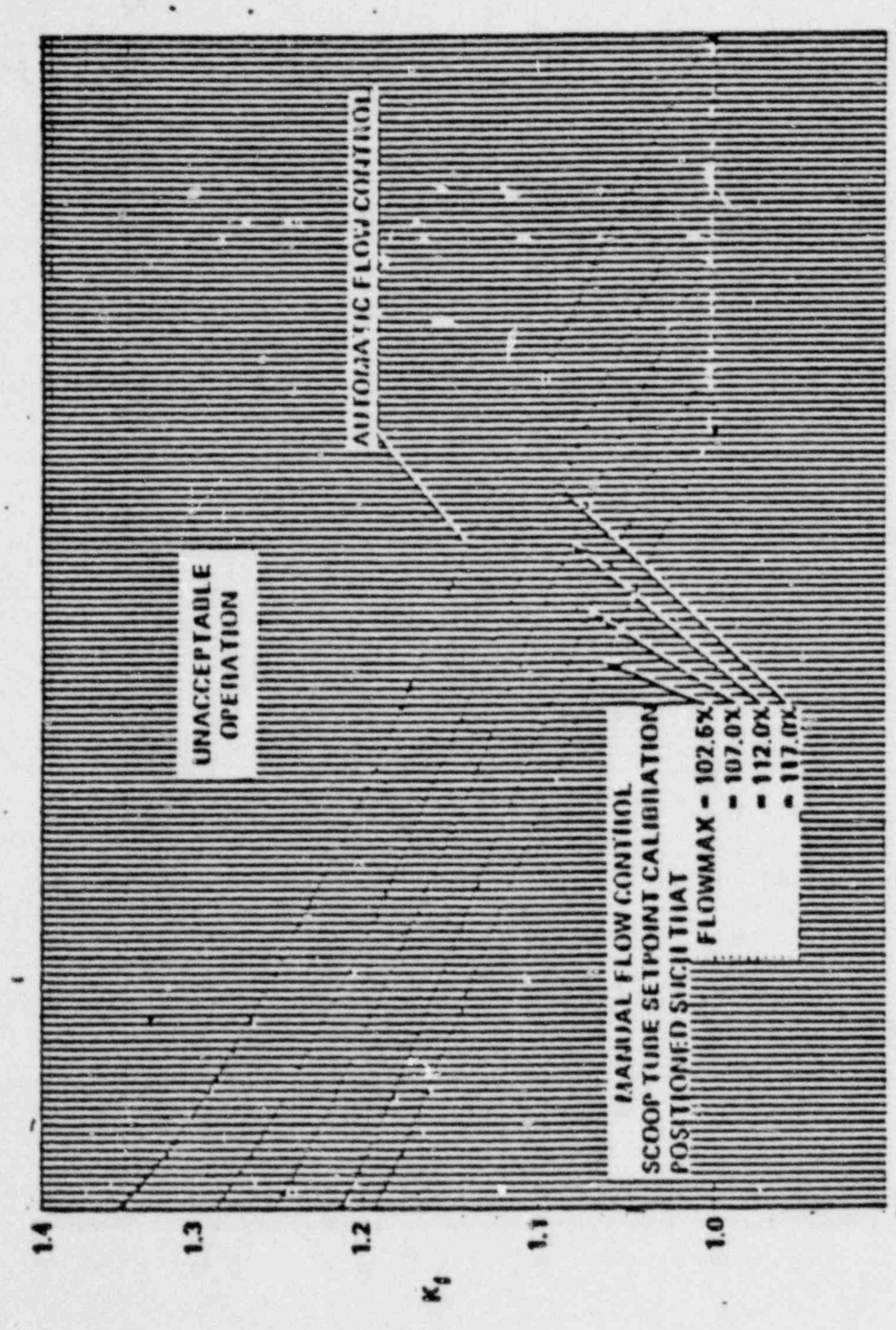
4.2.3 MCPR, as a function of core flow, shall be determined to be equal to or greater than the applicable limit determined from Figure 3.2.3-1:

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 with a LIMITING CONTROL ROD PATTERN for MCPR.

*Beginning of Cycle 3.
**End of Cycle 3.



CORE FLOW (%)

K_1 FACTOR

FIGURE 3.2.3-1

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 All LINEAR HEAT GENERATION RATES (LHGR's), shall not exceed 13.4 kw/ft.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% of RATED THERMAL POWER

ACTION:

With the LHGR of any fuel rod exceeding 13.4 kw/ft., 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 13.4 kw/ft:

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 3.3.4-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

NOTE

* When THERMAL POWER exceeds the preset power level of the RWM and RSCS.

a. The minimum number of OPERABLE CHANNELS may be reduced by one for up to 2 hours in one of the trip systems for maintenance and/or testing except for Rod Block Monitor function.

b. This function is bypassed if detector is reading > 100 cps or the IRM channels are on range 3 or higher.

c. This function is bypassed when the associated IRM channels are on range 8 or higher.

d. A total of 6 IRM instruments must be OPERABLE.

e. This function is bypassed when the IRM channels are on range 1.

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)	$\leq (0.66 W + 42\%) \frac{T^*}{MTPF}$	$\leq (0.66 W + 42\%) \frac{T^*}{MTPF}$
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	$\leq (0.66W + 41\%) \frac{T^*}{MTPF}$	$\leq (0.66 W + 41\%) \frac{T^*}{MTPF}$
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	$< 1 \times 10^5$ cps	$< 1 \times 10^5$ 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	< 108/125 of full scale	< 108/125 of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale

*T=2.43 for 8 x 8 fuel.
T=2.48 for 8 x 8 R fuel.
T=2.48 for P8x8R fuel.

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 clad 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, 3.2.1-5 and 3.2.1-6.

The calculational procedure used to establish the APLHGR shown on Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6 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 analyses 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, 3.2.1-5, and 3.2.1-6, (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; (4) The effects of core spray entrainment and counter-current 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.

Bases Table B 3.2.1-1

SI[illegible]IFICANT INPUTS PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

FOR BRUNSWICK-UNIT 1

Plant Parameters;

Core Thermal Power .... 2531 Mwt which corresponds 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 — 560

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**
All	8 x 8	13.4	1.4	1.2

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.

BASES

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity safety limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of 2.45 for 8 x 8 fuel and 2.48 for 8 x 8R and P8 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 2.45 for 8 x 8 fuel and 2.48 for 8 x 8R and P8 x 8R fuel. 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 MCPR's 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 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.

The limiting transient which determines the required steady state MCPR limit is the turbine trip with failure of the turbine by pass. 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 multi-channel 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.

BASES

MINIMUM CRITICAL POWER RATIO (Continued)

The evaluation of a given transient begins with the system initial parameters shown in Attachment 5 of Reference 6 that are input to a GE-core dynamic behavior transient computer program described in NEDO-10802(5). Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDO-20566(1). The principal result of this evaluation is the reduction in MCPR caused by the transient.

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

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

The K_f factor values shown in Figure 3.2.3-1 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of all reactor coolant system safety/relief valves shall be OPERABLE with lift settings within ± 1% of the following values.*#

4 Safety-relief valves @ 1105 psig.
4 Safety-relief valves @ 1115 psig.
3 Safety-relief valves @ 1125 psig.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety valve function of one safety/relief valve inoperable, restore the inoperable safety valve function of the valve to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the safety valve function of two safety/relief valves inoperable, restore the inoperable safety valve function of at least one of the valves to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the safety valve function of more than two safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 The safety valve function of each of the above required safety/relief valves shall be demonstrated OPERABLE by verifying that the bellows on the safety/relief valves have integrity, by instrumentation indication, at least once per 24 hours.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperature and pressure.

#From Spring, 1980 until the maintenance outage in Sept., 1980, the safety-relief valve lift settings shall be arranged such that each safety-relief valve pair has a minimum nominal lift setting differential of 20 psi and shall be within ± 1% of the following values:

2 Safety-relief valves @ 1095 psig
3 Safety-relief valves @ 1105 psig
3 Safety-relief valves @ 1115 psig
3 Safety-relief valves @ 1125 psig