DIST:						
	Docket File	NRC PDR	LPDR	ORB#2 Rdg	DEisenhut	ORAB
	SNorris	SMacKay	OELD	EJordan	TBarnhart-4	ACRS-10
	OPA, CMiles	NSIC	JTaylor	SECY	RDiggs	GRAY
						Xtra-5

Docket No. 50-325

JUN 28 1983

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

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 56 to Facility Operating License No. DPR-71 for the Brunswick Steam Electric Plant (BSEP) Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated May 2, 1983.

The amendment changes the Technical Specifications to establish revised operating limits for BSEP Unit 1 operation for fuel Cycle No. 4. The amendment also changes the control rod average scram insertion times to reflect an administrative change from percent of rod insertion to rod notch position.

A copy of the related Safety Evaluation is also enclosed.

Sincerely,

Original signed by
D. B. Vassallo

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

Enclosures:
1. Amendment No. 56 to DPR-71
2. Safety Evaluation

cc w/enclosures:
See next page

Provided No requests for hearing are received

8307130018 830628
PDR ADOCK 05000325
P PDR

DBV/PR

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:AD:OR	OELD	
SURNAME	SNorris	SMacKay	DVassallo	GLainas	M.KARMAN	
DATE	6/20/83	6/21/83	6/21/83	6/28/83	6/23/83	

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

cc:

Richard E. Jones, Esquire
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

George F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. Charles R. Dietz
Plant Manager
P. O. Box 458
Southport, North Carolina 28461

Mr. Franky Thomas, Chairman
Board of Commissioners
P. O. Box 249
Bolivia, North Carolina 28422

Mrs. Chrys Baggett
State Clearinghouse
Budget & Management
116 West Jones Street
Raleigh, North Carolina 27603

Dayne H. Browns, Chief
Radiation Protection Branch
Division of Facility Services
Department of Human Resources
P. O. Box 12200
Raleigh, North Carolina 27605

U. S. Environmental Protection Agency
Region IV Office
Regional Radiation Representative
345 Courtland Street, N. W.
Atlanta, Georgia 30308

Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 1057
Southport, North Carolina 28461

James P. O'Reilly
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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. 56
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 May 2, 1983 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 56, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8307130021 830628
PDR ADOCK 05000325
P PDR

3. This license amendment is effective as of the date of 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: June 28, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 56

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
(or delete appropriate side)	
I	I
II	II
IV	IV
1-4 thru 1-8	1-4 thru 1-9
3/4 1-6	3/4 1-6
3/4 1-7	3/4 1-7
3/4 2-1 thru 3/4 2-11	3/4 2-1 thru 3/4 2-16
3/4 3-42	3/4 3-42
B 3/4 2-1	B 3/4 2-1
B 3/4 2-3	B 3/4 2-3

INDEX

DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

ACTION....1-1
AVERAGE PLANAR EXPOSURE....1-1
AVERAGE PLANAR LINEAR HEAT GENERATION RATE....1-1
CHANNEL CALIBRATION....1-1
CHANNEL CHECK....1-1
CHANNEL FUNCTIONAL TEST....1-1
CORE ALTERATION....1-2
CRITICAL POWER RATIO....1-2
DOSE EQUIVALENT I-131....1-2
$\bar{E}$-AVERAGE DISINTEGRATION ENERGY....1-2
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME....1-2
FREQUENCY NOTATION....1-3
IDENTIFIED LEAKAGE....1-3
ISOLATION SYSTEM RESPONSE TIME....1-3
LIMITING CONTROL ROD PATTERN....1-3
LINEAR HEAT GENERATION RATE....1-3
LOGIC SYSTEM FUNCTIONAL TEST....1-3
MAXIMUM TOTAL PEAKING FACTOR....1-3
MINIMUM CRITICAL POWER RATIO....1-4
ODYN OPTION A....1-4
ODYN OPTION B....1-4
OPERABLE - OPERABILITY....1-4
OPERATIONAL CONDITION....1-4
PHYSICS TESTS....1-4

INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS (Continued) PAGE

PRESSURE BOUNDARY LEAKAGE.....1-4
PRIMARY CONTAINMENT INTEGRITY.....1-5
RATED THERMAL POWER.....1-5
REACTOR PROTECTION SYSTEM RESPONSE TIME.....1-5
REFERENCE LEVEL ZERO.....1-5
REPORTABLE OCCURRENCE.....1-5
ROD DENSITY.....1-6
SECONDARY CONTAINMENT INTEGRITY.....1-6
SHUTDOWN MARGIN.....1-6
SPIRAL RELOAD.....1-6
SPIRAL UNLOAD.....1-6
STAGGERED TEST BASIS.....1-7
THERMAL POWER.....1-7
TOTAL PEAKING FACTOR.....1-7
UNIDENTIFIED LEAKAGE.....1-7
FREQUENCY NOTATION, TABLE 1.1.....1-8
OPERATIONAL CONDITIONS, TABLE 1.2.....1-9

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.0 APPLICABILITY		3/4 0-1
3/4.1 REACTIVITY CONTROL SYSTEMS		
3/4.1.1	SHUTDOWN MARGIN	3/4 1-1
3/4.1.2	REACTIVITY ANOMALIES	3/4 1-2
3/4.1.3	CONTROL RODS	
	Control Rod Operability	3/4 1-3
	Control Rod Maximum Scram Insertion Times	3/4 1-5
	Control Rod Average Scram Insertion Times	3/4 1-6
	Four Control Rod Group Insertion Times	3/4 1-7
	Control Rod Scram Accumulators	3/4 1-8
	Control Rod Drive Coupling	3/4 1-9
	Control Rod Position Indication	3/4 1-11
3/4.1.4	CONTROL ROD PROGRAM CONTROLS	
	Rod Worth Minimizer	3/4 1-14
	Rod Sequence Control System	3/4 1-15
	Rod Block Monitor	3/4 1-17
3/4.1.5	STANDBY LIQUID CONTROL SYSTEM	3/4 1-18
3/4.2 POWER DISTRIBUTION LIMITS		
3/4.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE	3/4 2-1
3/4.2.2	APRM SETPOINTS	3/4 2-9
3/4.2.3	MINIMUM CRITICAL POWER RATIO	3/4 2-10
3/4.2.4	LINEAR HEAT GENERATION RATE	3/4 2-16

MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

ODYN OPTION A

ODYN OPTION A shall be analyses which refer to minimum critical power ratio limits which are determined using a transient analysis plus an analysis uncertainty penalty.

ODYN OPTION B

ODYN OPTION B shall be analyses which refer to minimum critical power ratio limits determined using a transient analysis which includes a requirement for 20% scram insertion times to reduce the analysis uncertainty penalty.

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 13 of the 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-isolatable 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

b. All equipment hatches are closed and sealed.

c. Each containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2436 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REFERENCE LEVEL ZERO

The REFERENCE LEVEL ZERO point is arbitrarily set at 367 inches above the vessel zero point. This REFERENCE LEVEL ZERO is approximately mid-point on the top fuel guide and is the single reference for all specifications of vessel water level.

REPORTABLE OCCURRENCE

A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.8.1.8 and 6.9.1.9.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All automatic reactor building ventilation system isolation valves or dampers are OPERABLE or secured in the isolated position,

b. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.1.

c. At least one door in each access to the reactor building is closed.

d. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor would be subcritical assuming that all control rods capable of insertion are fully inserted except for the analytically determined highest worth rod which is assumed to be fully withdrawn, and the reactor is in the shutdown condition, cold, 68°F, and Xenon free.

SPIRAL RELOAD

A SPIRAL RELOAD is the reverse of a SPIRAL UNLOAD. Except for two diagonal fuel bundles around each of the four SRMs, the fuel in the interior of the core, symmetric to the SRMs, is loaded first.

SPIRAL UNLOAD

A SPIRAL UNLOAD is a core unload performed by first removing the fuel from the outermost control cells (four bundles surrounding a control blade). Unloading continues in a spiral fashion by removing fuel from the outermost periphery to the interior of the core, symmetric about the SRMs, except for two diagonal fuel bundles around each of the four SRMs.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TOTAL PEAKING FACTOR

The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

TABLE 1.1
FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
N.A.	Not applicable.

TABLE 1.2
OPERATIONAL CONDITIONS

OPERATIONAL CONDITIONS	MODE SWITCH POSITIONS	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown	> 212°F
4. COLD SHUTDOWN	Shutdown	≤ 212°F
5. REFUELING*	Refuel**	≤ 212°F

*Reactor vessel head unbolted or removed and fuel in the vessel.***
**See Special Test Exception 3.10.3.
***See Special Test Exception 3.10.1.

CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITIONS FOR OPERATION

3.1.3.3 The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
46	0.31
36	1.05
26	1.82
6	3.37

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
46	0.33
36	1.12
26	1.93
6	3.58

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion times of control rods exceeding the above limits, operation may continue and the provisions of Specification 3.0.4 are not applicable provided:

a. The control rods with the slower than average scram insertion times are declared inoperable,

b. The requirements of Specification 3.1.3.1 are satisfied, and

c. The Surveillance Requirements of Specification 4.1.3.2.c are performed at least once per 92 days when operation is continued with three or more control rods with slow scram insertion times.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

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

a. During two recirculation loop operation, the limits are shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, or 3.2.1-7.

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, or 3.2.1-7, 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.

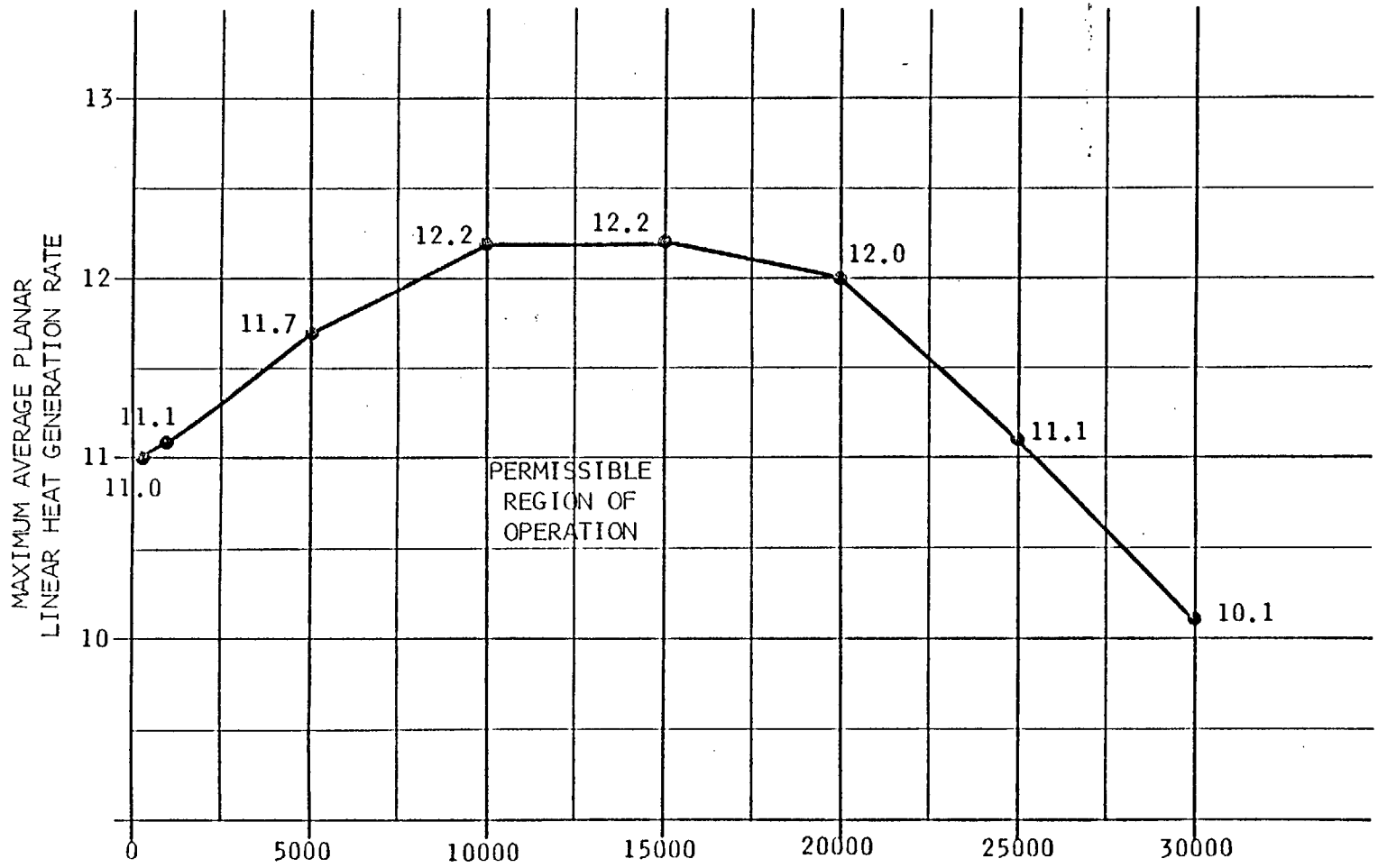
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the applicable limit determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, or 3.2.1-7:

a. At least once per 24 hours,

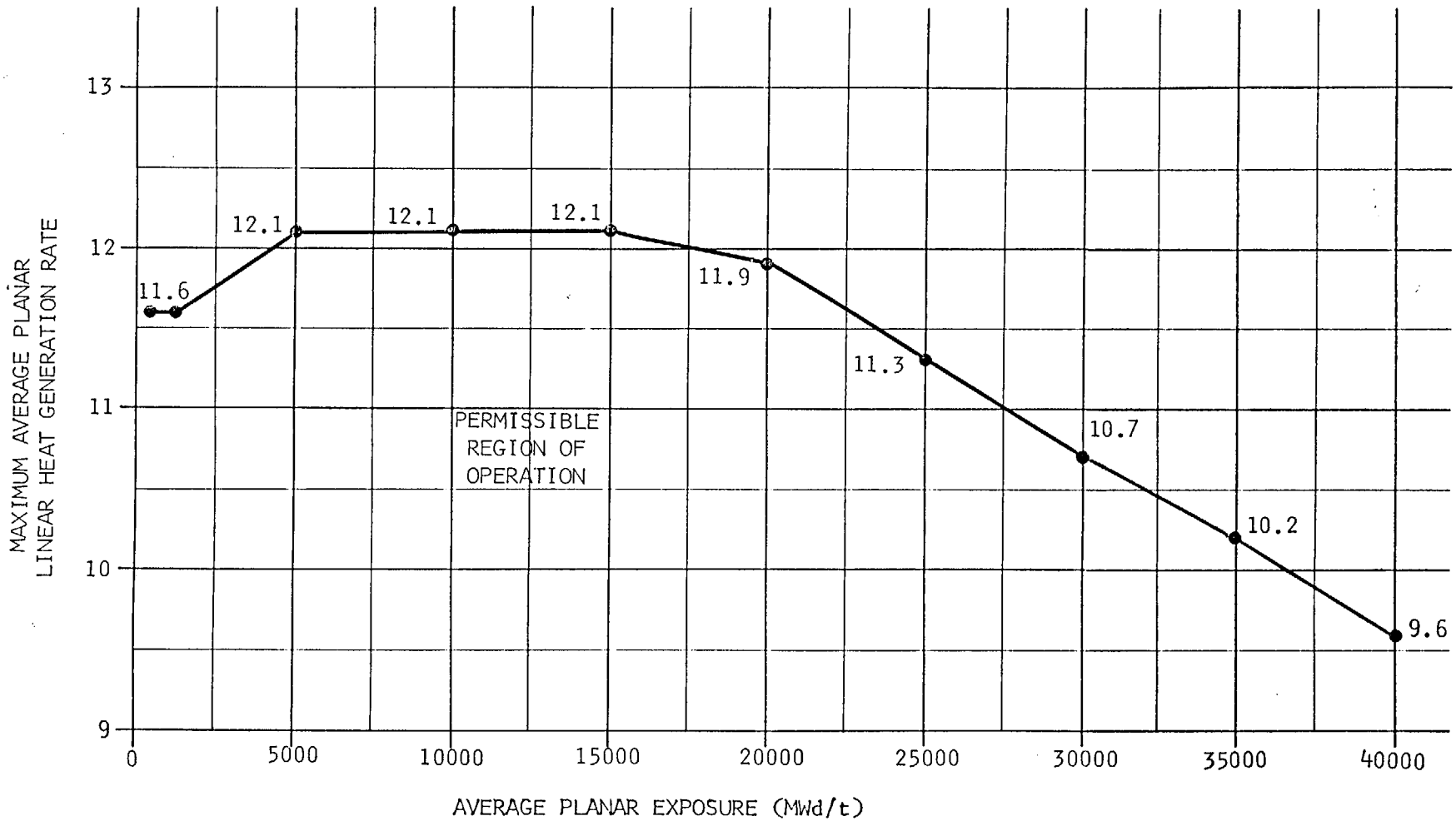
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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



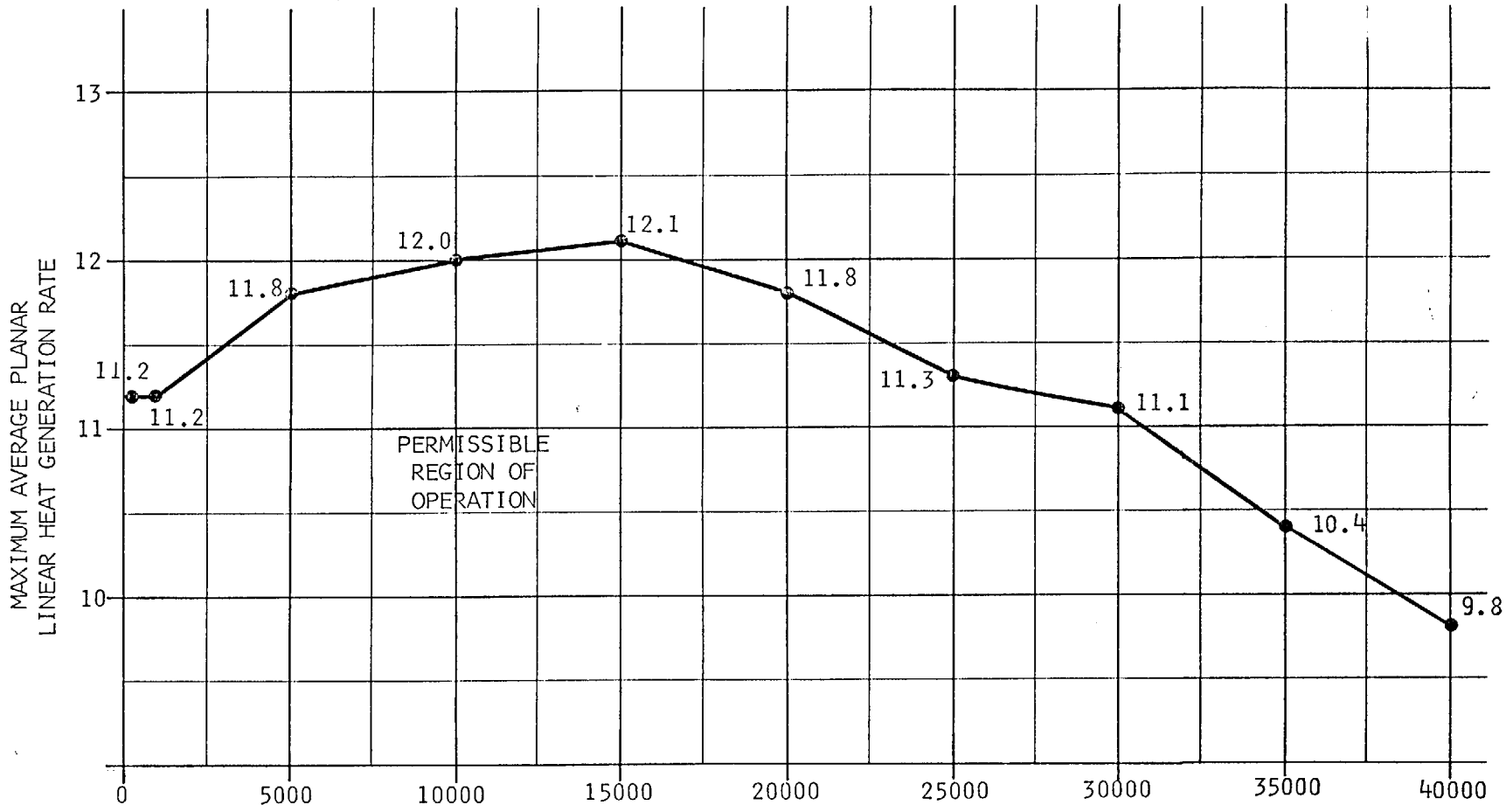
FUEL TYPE 2 (8 X 8)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-1



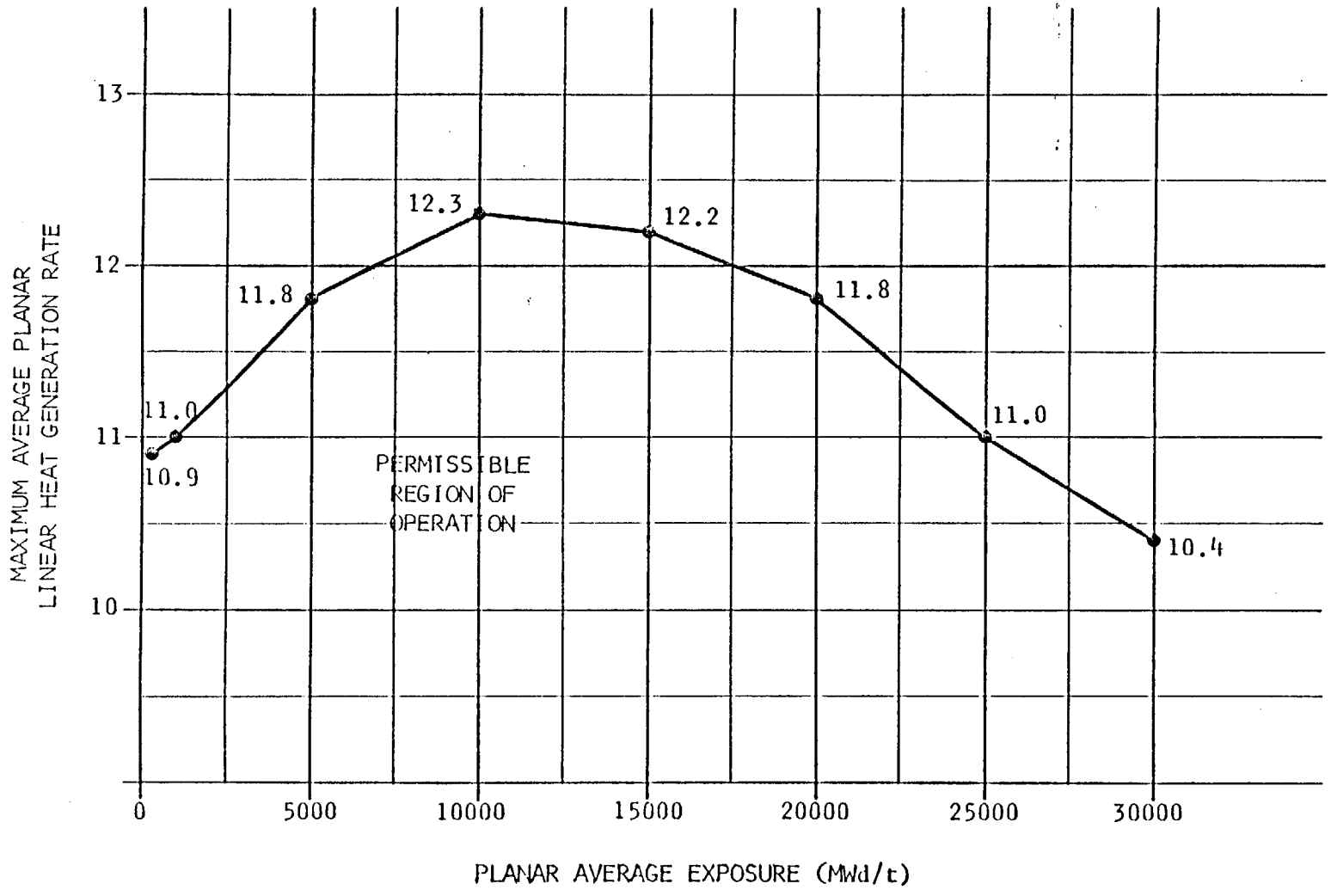
FUEL TYPE 8DRB265L (8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-2



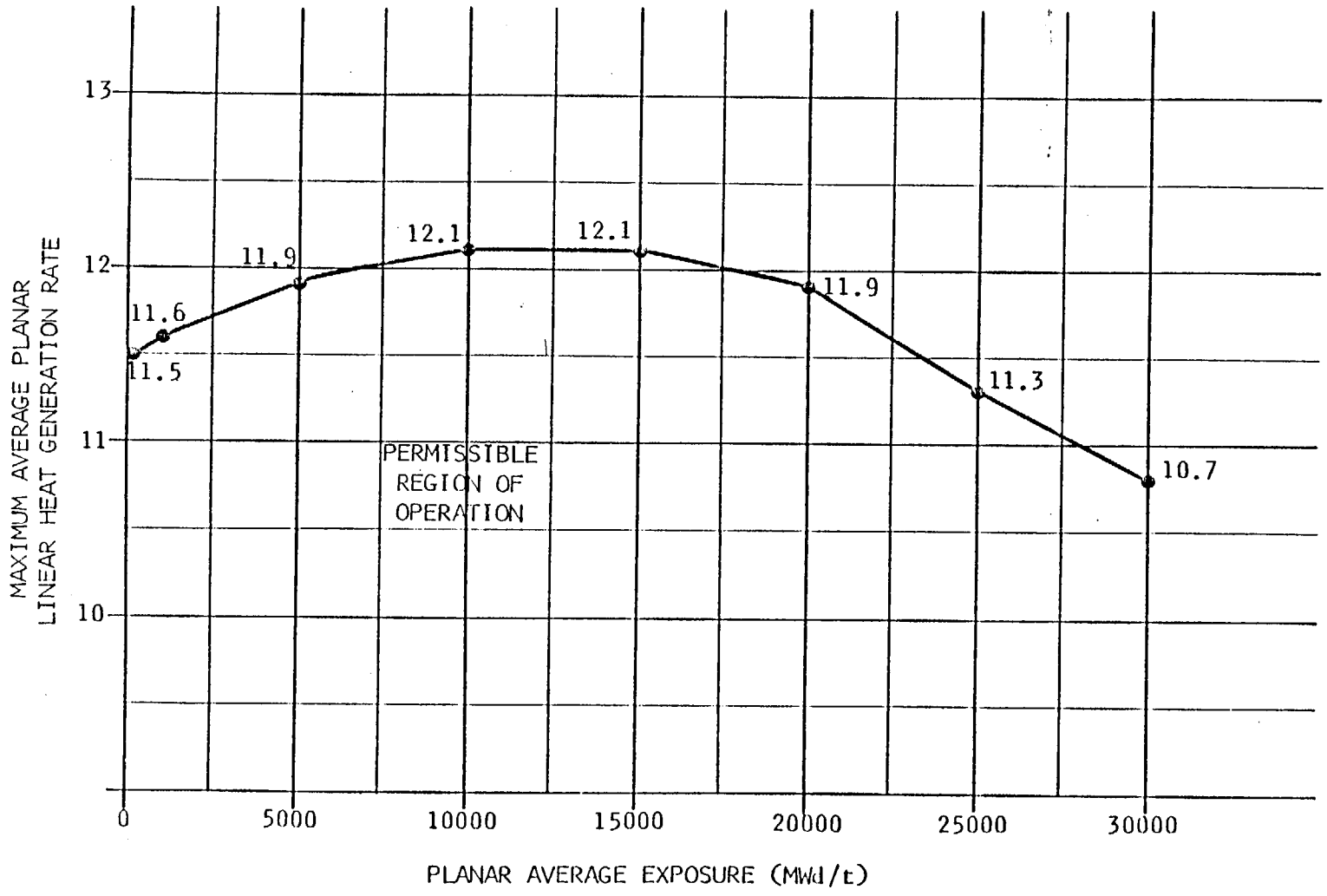
FUEL TYPE 8DRB283 (8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-3



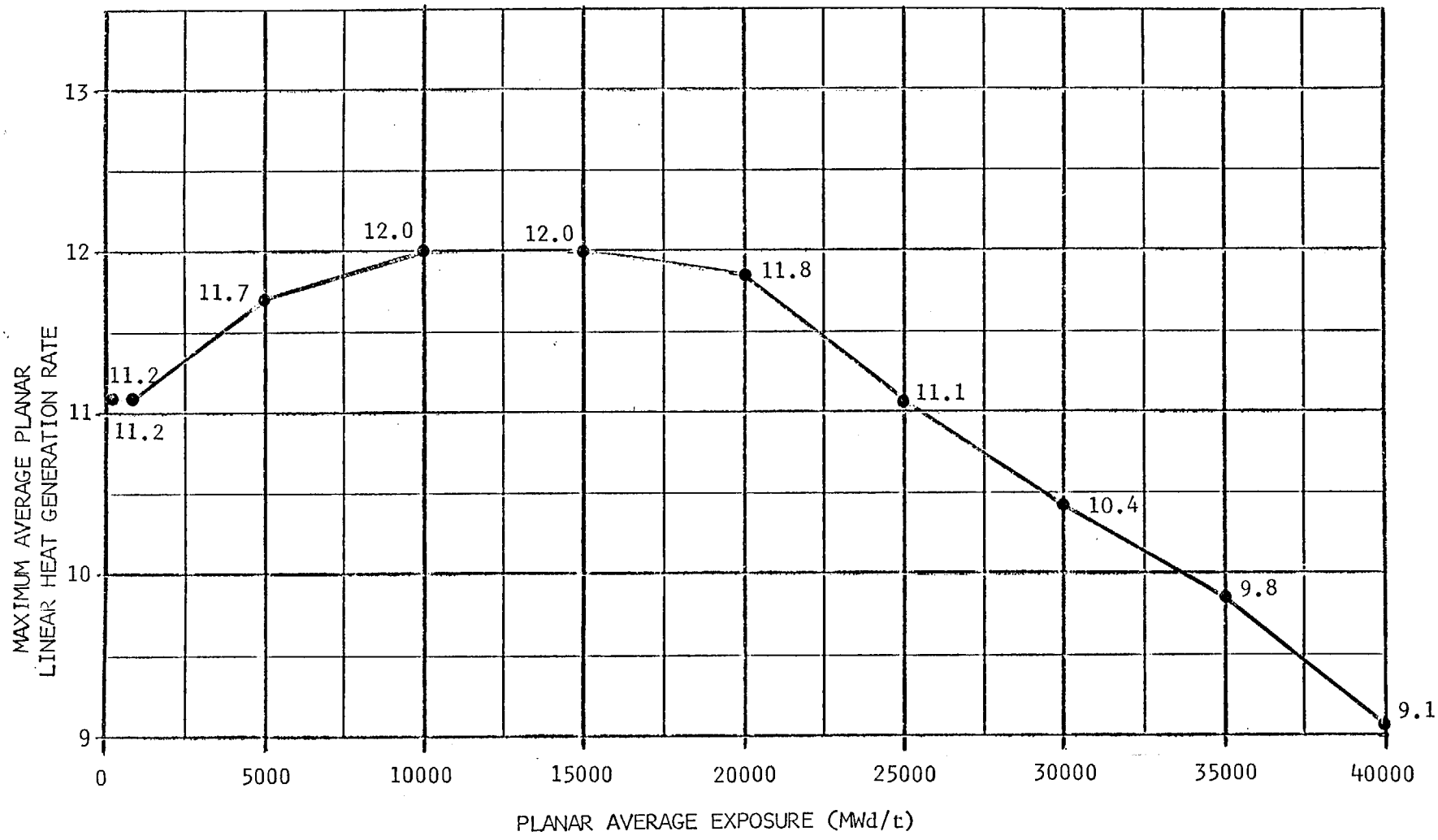
FUEL TYPE P8DRB285 (P8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-4



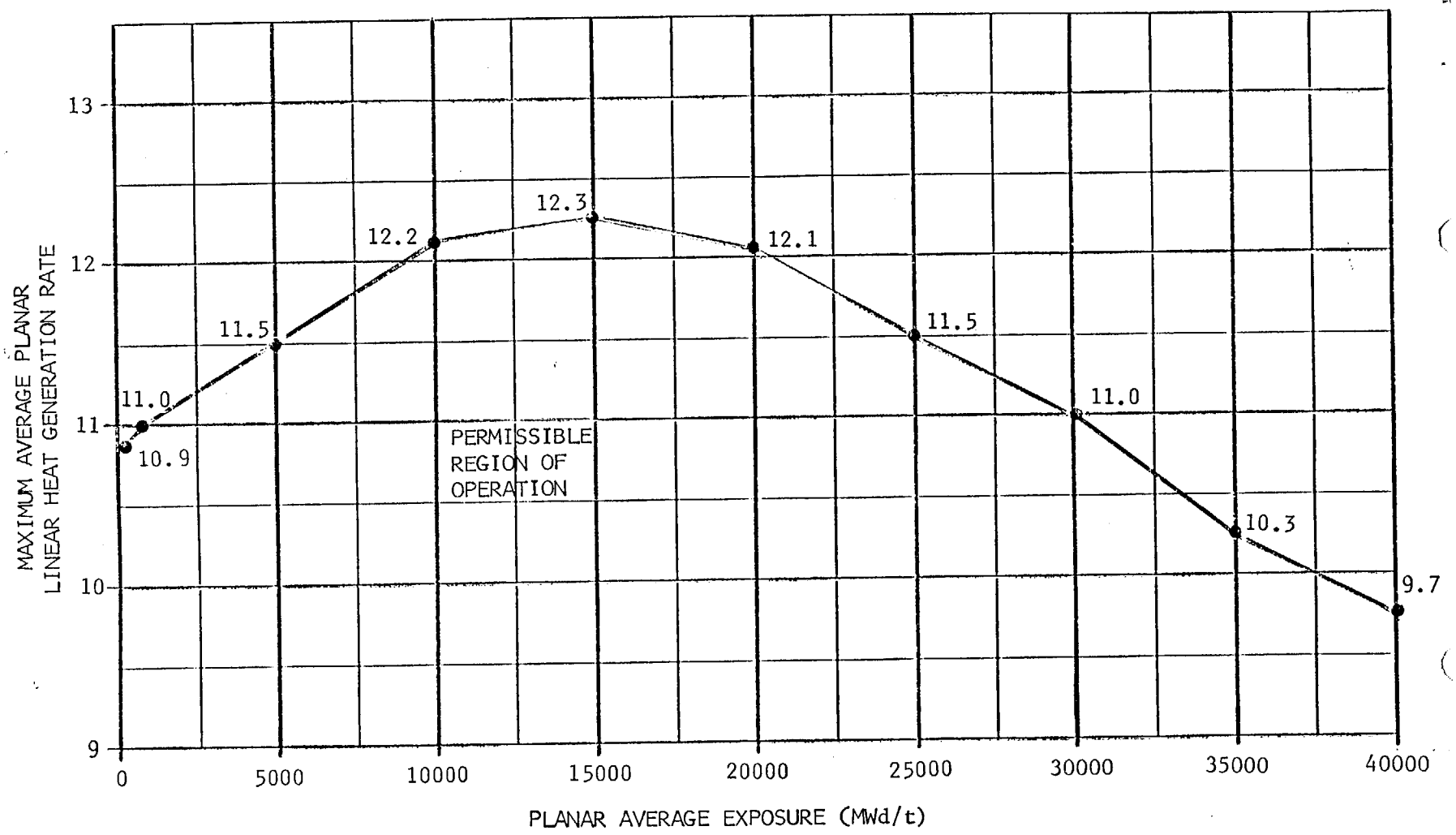
FUEL TYPE P8DRB265H (P8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-5



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

FIGURE 3.2.1-6



FUEL TYPE P8DRB299 (P8X8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-7

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

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, the MCPR limits are listed below:

   1. MCPR for 8 x 8 fuel = 1.26
   2. MCPR for 8 x 8R fuel = 1.27
   3. MCPR for P8 x 8R fuel = 1.28

b. EOC minus 2000 MWD/t to EOC with ODYN OPTION A analyses in effect, the MCPR limits are listed below:

   1. MCPR for 8 x 8 fuel = 1.37
   2. MCPR for 8 x 8R fuel = 1.38
   3. MCPR for P8 x 8R fuel = 1.41

c. BOC to EOC minus 2000 MWD/t with ODYN OPTION B analyses in effect, the MCPR limits are listed below:

   1. MCPR for 8 x 8 fuel = 1.21
   2. MCPR for 8 x 8R fuel = 1.25
   3. MCPR for P8 x 8R fuel = 1.25

d. EOC minus 2000 MWD/t to EOC with ODYN OPTION B analyses in effect, the MCPR limits are listed below:

   1. MCPR for 8 x 8 fuel = 1.26
   2. MCPR for 8 x 8R fuel = 1.27
   3. MCPR for P8 x 8R fuel = 1.29

APPLICABILITY: OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 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 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.

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

LIMITING CONDITION FOR OPERATION

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

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

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

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

LIMITING CONDITIONS FOR OPERATION (Continued)

ACTION:

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

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

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

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

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

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

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

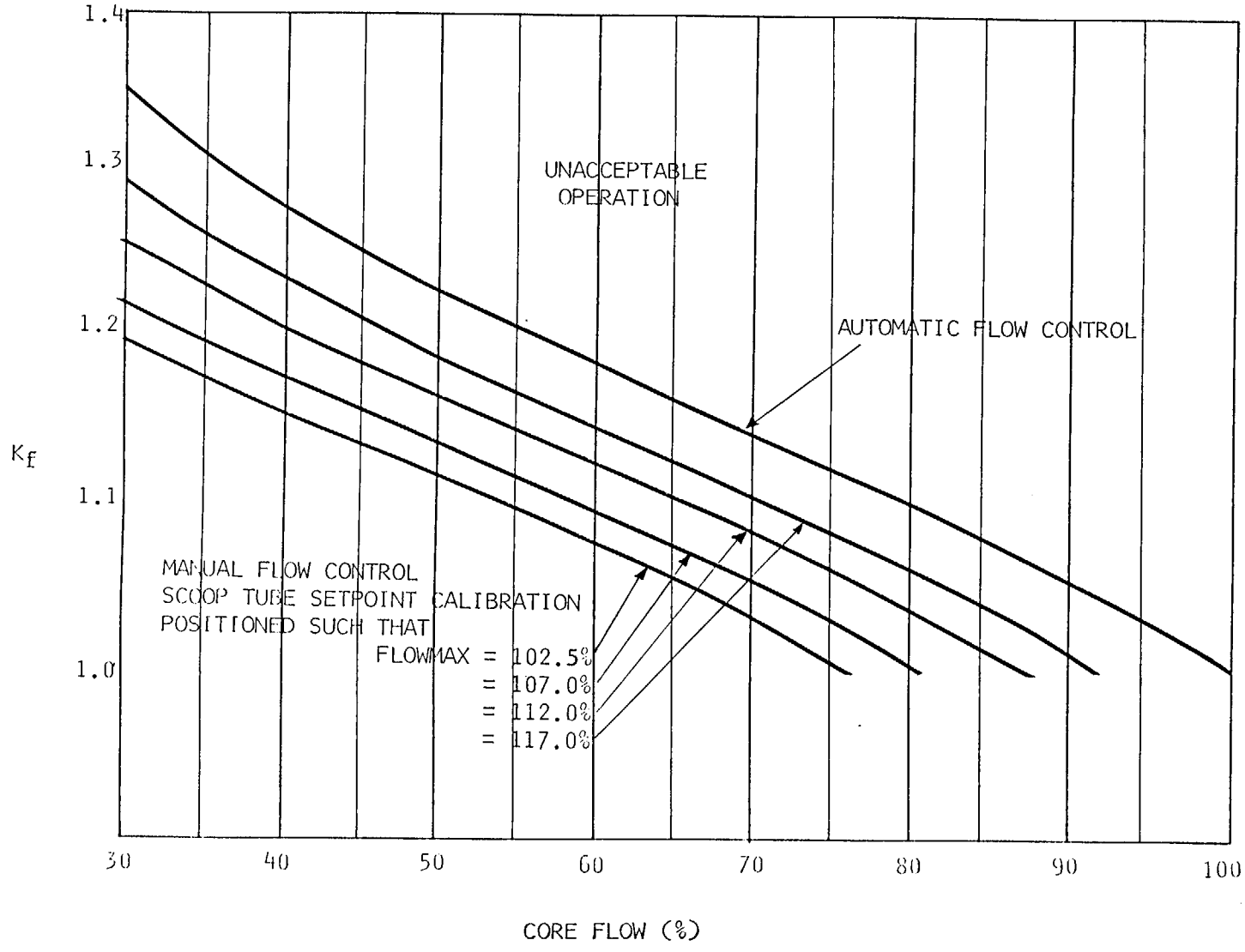
SURVEILLANCE REQUIREMENTS

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

TABLE 3.2.3.2-1

TRANSIENT OPERATING LIMIT MCPR VALUES

TRANSIENT	FUEL TYPE 8x8		8x8R		P8x8R	
NONPRESSURIZATION TRANSIENTS						
BOC → EOC	1.21		1.25		1.25	
TURBINE TRIP/LOAD REJECT WITHOUT BYPASS						
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
BOC → EOC - 2000	1.26	1.08	1.27	1.08	1.28	1.09
EOC - 2000 → EOC	1.37	1.25	1.38	1.26	1.41	1.29
FEEDWATER CONTROL FAILURE						
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
BOC → EOC - 2000	1.21	1.15	1.22	1.16	1.23	1.17
EOC - 2000 → EOC	1.33	1.26	1.34	1.27	1.36	1.29



K_f FACTOR

FIGURE 3.2.3-1

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, and P8 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 LHGR 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	$\leq (0.66W + 41\%)\frac{T^*}{MTPF}$	$\leq (0.66W + 41\%)\frac{T^*}{MTPF}$
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	$\leq 1 \times 10^5$ 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 (C11-LSH-N013E)		
a. Water Level High	$\leq$ 73 gallons	$\leq$ 73 gallons

*T=2.43 for 8x8 fuel
T=2.39 for 8x8R fuel
T=2.39 for P8x8R fuel

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 an assembly. The peak clad temperature is calculated assuming the 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 APLHGR 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, 3.2.1-6, or 3.2.1-7.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, and 3.2.1-7 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses performed with Reference 1 are: (1) The analysis assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, and 3.2.1-7 (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 countercurrent flow limitation as described in Reference 2, are included in the reflooding calculations.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of 2.43 for 8 x 8 fuel, 2.39 for 8 x 8R fuel, and 2.39 for 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 2.43 for 8 x 8 fuel, 2.39 for 8 x 8R fuel, and 2.39 for P8 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 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 bypass. This transient yields the largest 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 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. 56 TO FACILITY LICENSE NO. DPR-71

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

DOCKET NO. 50-325

1.0 Introduction

By letter dated May 2, 1983 (Ref. 1) Carolina Power and Light Company (the licensee) proposed to change the Technical Specifications for the Brunswick Steam Electric Station Unit 1 to permit its operation for fuel Cycle 4. In support of the application the licensee submitted a Supplemental Reload licensing document (Ref. 2) and a commitment to add certain technical Specifications with regard to the turbine bypass and high reactor water level trip systems (Ref. 9). We have reviewed the licensee's application and prepared the following evaluation.

2.0 Evaluation

2.1 Fuel System Design

The licensee's analysis of the safety considerations involved in the proposed fourth cycle of operation at Brunswick Unit 1 are described in the Supplemental Reload Licensing Submittal (Ref. 2). In all fuel-design-related areas, the reload submittal relies on the General Electric generic report, Generic Reload Fuel Application (Ref. 3). Reference 3 has been reviewed and approved by the staff (Ref. 10). With the exception of the revised Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits, we conclude that additional staff review of those portions of the generic application concerning the standard fuel designs is unnecessary for the Cycle 4 application.

The licensee's submittal provided MAPLHGR limits for the standard 8x8, 8x8R and P8x8R fuel assemblies in the Cycle 4 core. Although the methodology used in Reference 3 is generically applicable for the MAPLHGR limit determination, the staff believes that the effects of enhanced fission gas release at high burnup (i.e., greater than 20 MWd/kg/U) were not adequately considered in the fuel performance model. In response to this concern, General Electric (GE) requested (Refs. 5 and 6) 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

8307130025 830628
PDR ADOCK 05000325
P PDR

acceptable provided that certain plant-specific conditions were met (Ref. 7). The licensee has stated (Ref. 8) that the GE proposal is applicable to the Brunswick Unit 1 safety analysis. On this basis we conclude that the proposed MAPLHGR limits are appropriate for Cycle 4 operation.

2.2 Nuclear Design

The nuclear design and analysis of the proposed reload has been performed by the methods described in Reference 3. This report has been approved for use in the design and analysis of reloads in BWR reactors and its use is acceptable for this reload. The results of the nuclear design analysis are consistent with those for similar reloads and are acceptable.

2.3 Thermal Hydraulic Design

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

The review includes the following areas: (1) safety limit minimum critical power ratio (MCPR), (2) operating limit MCPR and (3) thermal-hydraulic stability.

The licensee has submitted the reload analysis report for Cycle 4 operation (Ref. 2), which is based on the approved GE report (Ref. 3).

Discussion of the review concerning the thermal-hydraulic design for Cycle 4 operation follows:

Safety Limit MCPR

A safety limit MCPR has been established to assure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during normal operation and anticipated transients. The approved safety limit MCPR of 1.07 as sated in Ref. 3 was used for the Cycle 4 analyses.

Operating Limit MCPR (OLMCPR)

Various transients could reduce MCPR below the intended safety limit MCPR during Cycle 4 operation. The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the critical power ratio (ΔCPR). The ΔCPR values given in Section 9 of Ref. 2 are plant specific values calculated by using the methods given in Ref. 3 including ODYN methods. The calculated ΔCPRs were adjusted to reflect "A" Option or "B" Option ΔCPRs by employing the conversion method described in Ref. 4. The cycle MCPR values are determined by adding the ΔCPRs to the safety limit MCPR. Section 11 of Ref. 2 presents both the cycle MCPR values for the pressurization and non-pressurization events. The maximum cycle MCPR values (for "A" Option and "B" Option) in Section 11 are specified as the operating limit MCPRs and are incorporated into the Technical Specifications.

The OLMCPR must be >1.21 for 8x8 fuel and >1.25 for 8x8R and P8x8R fuel types at fuel exposures from Begining of Cycle (BOC) to End of Cycle (EOC) minus 2000 MWD/t. For fuel exposures from EOC minus 2000 MWD/t to EOC, the OLMCPR must be >1.26 for 8x8 fuel, >1.27 for 8x8R fuel and >1.29 for P8x8R fuel.

Thermal-Hydraulic Stability

The results of thermal-hydraulic analyses (Ref. 2) show that the maximum core stability decay ratio is 0.72 for Cycle 4 as compared to 0.74 for Cycle 3.

We find that (1) the calculated decay ratio for Cycle 4 is less than that for the approved Cycle 3 operation and (2) operation for more that 12 hours in the natural circulation mode will be prohibited by Technical Specification 3/4.4.1. We therefore conclude that the thermal-hydraulic stability results are acceptable for Cycle 4 operation.

2.4 Transient and Accident Analyses

The 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 (280 calories per gram) for this event and is therefore acceptable. The fuel misorientation event is not limiting at any time in the cycle. The rod withdrawal event is the limiting event with respect to OLMCPR during the portion of the cycle prior to 2000 MWD/t before End-of-Cycle if the ODYN Code OPTION B Transient Analysis is used.

Since approved methods have been used to perform the analyses and to obtain input parameters for them, we conclude that the transient analyses of the three cited events is acceptable. For discussion of core-wide transient analyses, see OLMCPR above.

2.5 Technical Specification Changes

The control rod average scram insertion times were changed to correct errors that were introduced when the designation of control rod position was changed from percent of rod insertion to rod notch position. The designation was changed in order to conform to the BWR Standard Technical Specifications but the conversion of the positions and times was not accurate. The new values agree with the original bases for insertion times and are therefore, acceptable.

Changes were made in Technical Specifications 3/4.2.2 and 3/4.2.4 in order to establish the operating limits on the flow-biased APRM scram set point based on new values of the Design Total Peaking Factor. These new values are consistent with the previously established linear heat generation rate (13.4 KW/Ft) and are therefore, acceptable. In addition, the editorial changes are correct.

Section 3/4.2.3 and Table 3.2.3.2-1 of the Technical Specifications have been modified to include the operating limit MCPR for Cycle 4 operation. We find that for the limiting event for "B" Option, feedwater controller failure that causes maximum demand, credit is assumed for operation of the high water level (L8) trip and turbine bypass system. Accordingly, we require that Technical Specifications be included to ensure the operability of these systems.

To assure an acceptable level of performance, our position is that this equipment (the turbine bypass system and the level 8 high water level trip) should be identified in the Technical Specifications with limiting conditions for operation, surveillance requirements and with adequate degree of power reduction, in case of inoperability. The licensee has agreed to submit, within 30 days of startup of Cycle 4 operation, the proposed changes to the Technical Specifications that will incorporate the Limiting Conditions for Operation and surveillance requirements for the turbine bypass and high water level trip systems (Ref. 9). Based on our review of the significance of these trip systems with regard to the limiting transients during this fuel cycle, we find this commitment acceptable.

3.0 Summary of Evaluation

The following statements summarize the staff conclusions concerning the Cycle 4 reload for Brunswick Unit 1.

Based on our previous generic approval of the fuel-related design features of BWR reloads provided by General Electric, we find these features to be acceptable for Brunswick Unit 1 Cycle 4. On the basis that certain plant specific conditions relating to extended fuel burnup have been met, we conclude that extension of the MAPLHGR limits to higher burnup values is acceptable for Cycle 4.

In the course of our review, we found that approved methods were used and that the results are consistent with those for similar reloads. We therefore, conclude that the nuclear and thermal-hydraulic design of the reload is acceptable. On the same basis we conclude that the transient and accident analyses are acceptable and that the results support the operation of Brunswick Unit 1 for Cycle 4.

We further conclude that the proposed changes in the plant Technical Specifications are acceptable.

In summary, we conclude that Brunswick Unit 1 may be operated for Cycle 4 without undue hazard to the public health and safety.

3.1 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

3.2 Final Determination of No Significant Hazards Consideration

On May 19, 1983 the Commission published a notice in the Federal Register (48 FR 22658) seeking public comment on its proposed determination that this amendment involves no significant hazards consideration. No public comments were received. The state of North Carolina was consulted on this matter and had no comments on the proposed determination. The Commission has provided examples of amendments that are considered not likely to involve significant hazards considerations (48 CFR 14870). One such amendment is a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the Commission for a previous core at the facility in question are involved.

In reviewing the application for this amendment, we found that this refueling of the Brunswick Steam Electric Plant, Unit 1 involves no fuel assemblies significantly different from those used in a previous refueling and found acceptable to the Commission. Furthermore, we found that there are no significant changes to the acceptance criteria for the technical specifications. As discussed above, the analytical methods used to establish the changes in the technical specifications and demonstrate conformance with the regulations have been found previously acceptable to the Commission. None of the changes set forth in this amendment involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety. Therefore, the Commission has made a final determination that this amendment does not involve a significant hazards consideration.

4.0 Conclusion

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

Dated: June 28, 1983

Principal Contributors: J. Voglewede
W. Brooks
S. Sun

References

1. Letter from P. W. Howe (CP&L) to D. Vassallo (NRC), Request for Revision to Technical Specifications (Fuel Cycle No. 4-Reload Licensing), May 2, 1983.

2. Y1003J01A53, Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 1, Reload 3 (Without Recirculation Pump Trip), January 1983, submitted by letter dated May 6, 1983 from S. R. Zimmerman (CP&L) to D. B. Vassallo (NRC).

3. NEDE-24011-P-A-4, General Electric Boiling Water Reactor Generic Reload Fuel Applications, January 1982.

4. Letter from R. Bucholz (GE) to P. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model", September 5, 1980.

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

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

7. 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.

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

9. P. W. Howe (CP&L) letter to D. B. Vassallo (NRC) dated June 20, 1983.

10. D. G. Eisenhut (NRC) letter to R. Gridley (GE) May 12, 1978