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Dockets Nos. 50-325
and 50-324

Mr. J. A. Jones
Executive Vice President
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. DPR-71 and Amendment No. 48 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant (BSEP), Units Nos. 1 and 2, respectively.

The amendment for Unit No. 1 provides Technical Specifications for the protective instrumentation associated with the Anticipated Transient Without Scram (ATWS) Recirculation Pump Trip (RPT). These specifications were inadvertently omitted in converting from custom to standard Technical Specifications (Reference Amendment No. 12 to DPR-71 dated November 23, 1977). These specifications have been discussed with and agreed to by your staff.

The amendment for Unit No. 2 also provides Technical Specifications for the ATWS RPT (Reference Amendment No. 39 to DPR-62 dated November 23, 1977). These specifications have been discussed with and agreed to by your staff.

In addition, the amendment for Unit No. 2 consists of changes to the Technical Specifications in response to your request dated February 2, 1979, as supplemented March 16, March 21, March 27, April 13, April 27, and May 1, 1979. This amendment changes the Technical Specifications for Unit 2 to establish revised safety and operating limits for operation in Cycle 3 with 7x7, 8x8 and 8x8R fuel, and permits operation of BSEP Unit No. 2 in Cycle 3 following the current refueling outage. This amendment does not include credit for the end-of-cycle recirculation pump trip (EOC RPT) feature which was installed in Unit No. 2 during the refueling outage.

CP 1

7906070386

OFFICE➤						
SURNAME➤						
DATE➤						

NRC FORM 318 (9-76) NRCM 0240

U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 24 to DPR-71
2. Amendment No. 48 to DPR-62
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

S Lewis has seen SE & agrees with notice TAI 5/2

Concur subject to review of SE to determine whether past notice was appropriate

discussed with Grimes. TAI 5/2

OFFICE	ORB #3	ORB #3	AD:E&P	OELD	ORB #3
SURNAME	SSheppard	JHannon:mjf	~~BGrimes~~	SHLewis	TIppolito
DATE	5/2/79	5/2/79	~~/ /79~~	5/2/79	5/2/79

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

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Mr. Steve J. Varnam
Chairman, Board of County
Commissioners of Brunswick County
Southport, North Carolina 28461

Denny McGuire (Ms)
State Clearinghouse
Division of Policy Development
116 West Jones Street
Raleigh, North Carolina 27603

Southport - Brunswick County Library
109 W. Moore Street
Southport, North Carolina 28461

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
US EPA
Crystal Mall #2
Arlington, Virginia 20460

U.S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, NW
Atlanta, Georgia 30308

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. 24
License No. DPR-71

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

   A. The facility will operate in conformity with the provisions of the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission;

   B. 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;

   C. 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

   D. 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.C(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 24, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

7906070391

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 2, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 24

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
V	V
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IX*	IX*
X	X
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	3/4 3-62
	3/4 3-63
	3/4 3-64
	3/4 3-65
B 3/4 3-3*	B 3/4 3-3*
B 3/4 3-4	B 3/4 3-4

*Overleaf pages supplied for convenience

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TABLE 3.3.5.7-1 (Continued)

INSTRUMENT LOCATION		MINIMUM INSTRUMENTS OPERABLE		
		FLAME	HEAT	SMOKE
4. Service Water Building				
Zone 1	4'	0	0	6
Zone 2	20	0	0	5
5. AOG Building				
Zone 1	20'	1	0	0
Zone 2	20'	1	0	0
Zone 3	20'	1	5	1
Zone 4	37' - 49'	1	6	0

3/4.3.6 ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.1 The Anticipated Transient Without Scram recirculation pump trip (ATWS-RPT) system instrumentation trip systems shown in Table 3.3.6.1-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.1-2.

APPLICABILITY: CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6.1-2, declare the trip system inoperable until the trip system is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE trip systems per operating pump not satisfied for one Trip Function, restore the inoperable trip system to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.3.6.1.1 Each ATWS recirculation pump trip system instrumentation trip system shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.1.1-1.

4.3.6.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

TABLE 3.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE TRIP SYSTEMS PER OPERATING PUMP
1. Reactor Vessel Water Level - Low Low, Level 2 (B21-LIS-N024 A, B; B21-LIS-N025 A, B)	1
2. Reactor Vessel Pressure-Low (B21-PS-N045 A, B, C, D)	1

TABLE 3.3.6.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low low, Level 2 (B21-LIS-N024 A, B; B21-LIS-N025 A, B)	$\geq$ -38 inches	$\geq$ -38 inches
2. Reactor Vessel Pressure-Low (B21-PS-N045 A, B, C, D)	$\geq$1120 psig	$\geq$1120 psig

TABLE 4.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2 (B21-LIS-N024 A, B; B21-LIS-N025 A, B)	S	M	R
2. Reactor Vessel Pressure - Low (B21-PS-N045 A, B, C, D)	NA	M	R

MONITORING INSTRUMENTATION (Continued)

3/4.3.5.2 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of CFR 50.

3/4.3.5.3 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97 "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975.

3/4.3.5.4 SOURCE RANGE MONITORS

The source range monitors provide the operator with information on the status of the neutron level in the core at very low power levels during startup. At these power levels reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adaquate information is available without the SRM's and they can be retracted.

3/4.3.5.5 CHLORINE DETECTION SYSTEM

The OPERABILITY of the chlorine detection systems ensures that an accidential chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personel. Upon detection of a high concentration of chlorine the control room emergency ventilation system will automatically isolate the control room and initiate operation in the recirculation mode to provide the required protection. The detection systems required by this specifications are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an accidental Chlorine Release.

MONITORING INSTRUMENTATION (Continued)

3/4.3.5.6 CHLORIDE INTRUSTION MONITORS

The chloride intrusion monitors provide adequate warning of any leakage in the condenser or hotwell so that actions can be taken to mitigate the consequences of such intrusion in the reactor coolant system. With only a minimum number of instruments available increased sampling frequency provides adequate information for the same purpose.

3/4.3.5.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.6 ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

The ATWS recirculation pump trip system has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report NEDO-10349, dated March, 1971.

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-62

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

   A. The applications for amendments by Carolina Power & Light Company (the licensee) dated February 2, 1979, as supplemented March 16, 21 and 27, April 13 and 27, and May 1, 1979, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

2.C(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 2, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

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 page is reflected by a marginal line.

Remove	Insert
III*	III*
IV	IV
V	V
VI*	VI*
IX*	IX*
X	X
2-1	2-1
2-2*	2-2*
B 2-1	B 2-1
B 2-2*	B 2-2*
B 2-9	B 2-9
B 2-10*	B 2-10*
3/4 1-17	3/4 1-17
3/4 1-18*	3/4 1-18*
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-11	3/4 2-11
3/4 2-12*	3/4 2-12*
3/4 2-13	3/4 2-13
3/4 3-39	3/4 3-39
3/4 3-40*	3/4 3-40*
3/4 3-41*	3/4 3-41*
3/4 3-42	3/4 3-42
3/4 3-43	3/4 3-43
3/4 3-44*	3/4 3-44*

Remove	Insert
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	3/4 3-62
	3/4 3-63
	3/4 3-64
	3/4 3-65
B 3/4 1-1*	B 3/4 1-1*
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4*	B 3/4 2-4*
B 3/4 2-5	B 3/4 2-5
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*Overleaf pages supplied for convenience

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER (Low Pressure or Low Flow)

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 800 psia or core flow less than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 800 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

THERMAL POWER (High Pressure and High Flow)

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than 800 psia and core flow greater than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 and the reactor vessel steam dome pressure greater than 800 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure $\leq$ 1325 psig within 2 hours.

2.1 SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: CONDITIONS 3, 4 and 5

ACTION:

With the reactor water level at or below the top of the active irradiated fuel, manually initiate the low pressure ECCS to restore the reactor vessel water level, after depressurizing the reactor vessel, if required.

2.1 SAFETY LIMITS

BASES

2.0 The fuel cladding, reactor pressure vessel and primary sys. piping are the principal barriers to the release of radioactive materials to the environs. Safety limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MINIMUM CRITICAL POWER RATIO (MCPR) is no less than 1.07. MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER (Low Pressure or Low Flow)

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 800 psia or core flows less than 10% of rated flow. Therefore the fuel cladding integrity limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 800 psia is conservative.

2.1.2 THERMAL POWER (High Pressure and High Flow)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power, result in an uncertainty in the value of the critical power. Therefore the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB,[1] which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), (GEXL), correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia
Mass Flux:	0.1 to 1.25 10^6 lb/hr-ft^2
Inlet Subcooling:	0 to 100 Btu/lb
Local Peaking:	1.61 at a corner rod to
	1.47 at an interior rod

Reference

1 "General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design application," NEDO-10958 and NEDO-10958.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Setpoints specified in Table 2.2.1-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. Range 10 allows the IRM instruments to remain on scale at higher power levels to provide for additional overlap and also permits calibration at these higher powers.

The most significant source of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed, Section 7.5 of the FSAR. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM's are not yet on scale. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 1% of RATED THERMAL POWER, thus maintaining MCPR above 1.07. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. This margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup, is not much colder than that already in the system, temperature coefficients are small and control rod patterns are constrained by the RSCS and RWM. Of all

2. Average Power Range Monitor (Continued)

the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% APRM trip remains active until the mode switch is placed in the Run position.

The APRM flow biased trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation; i.e., the thermal power of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer. Analyses demonstrate that with only the 120% trip setting, none of the abnormal operational transients analyzed violates the fuel safety limit and there is substantial margin from fuel damage. Therefore the use of the flow referenced trip setpoint, with the 120% fixed setpoint as backup, provides adequate margins of safety.

The APRM trip setpoint was selected to provide adequate margin for the Safety Limits and yet allows operating margin that reduces the possibility of unnecessary shutdowns. The flow referenced trip setpoint must be adjusted by the specified formula in order to maintain these margins.

3. Reactor Vessel Steam Dome Pressure-High

High Pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating, will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux counteracting the pressure increase by decreasing heat generation. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both Rod Block Monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: CONDITION 1, when THERMAL POWER is greater than the preset power level of the RWM and RSCS

ACTION:

a. With one RBM channel inoperable, POWER OPERATION may continue provided that either:

   1. The inoperable RBM channel is restored to OPERABLE status within 24 hours, or

   2. The redundant RBM is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable RBM is restored to OPERABLE status, and the inoperable RBM is restored to OPERABLE status within 7 days, or

   3. THERMAL POWER is limited such that MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of any single control rod that is capable of withdrawal.

   Otherwise, trip at least one rod block monitor channel.

b. With both RBM channels inoperable, trip at least one rod block monitor channel within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and during the OPERATIONAL CONDITIONS specified in Table 4.3.4-1.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE with:

a. An OPERABLE flow path from the storage tank to the reactor core, containing two pumps and and two inline explosive injection valves,

b. The contained solution volume-concentration within the limits of Figure 3.1.5-1, and

c. The solution temperature above the limit of Figure 3.1.5-2.

APPLICABILITY: CONDITIONS 1, 2, and 5.

ACTION:

a. In CONDITION 1 or 2:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the standby liquid control system inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In CONDITION 5:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 31 days or suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

2. With the standby liquid control system inoperable, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

3. The provisions of Specification 3.0.3 are not applicable.

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, 3.2.1-6 or 3.2.1-7.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 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.

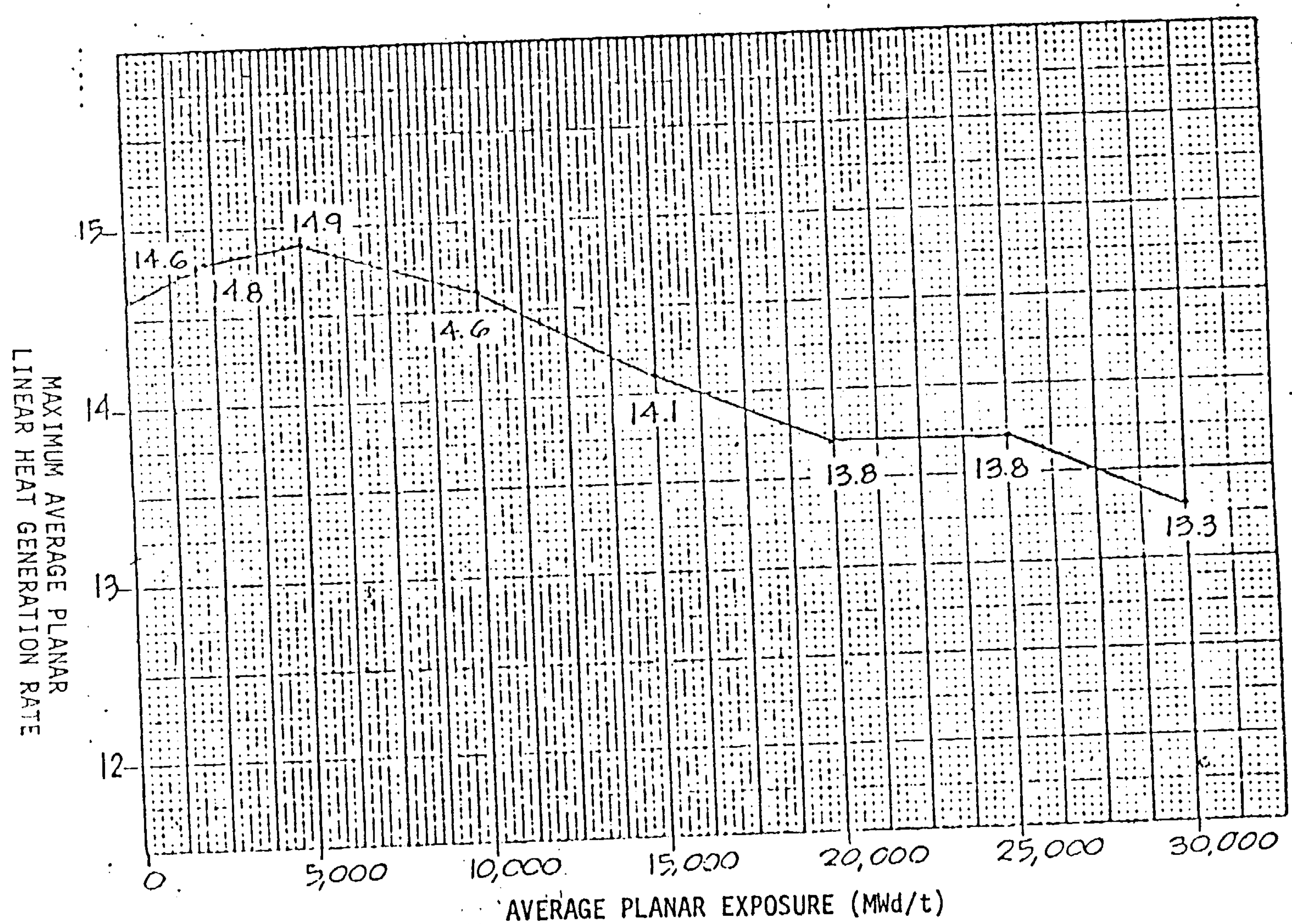
SURVEILLANCE REQUIREMENTS

4.2.1 All 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, 3.2.1-6 or 3.2.1-7:

a. At least once per 24 hours,

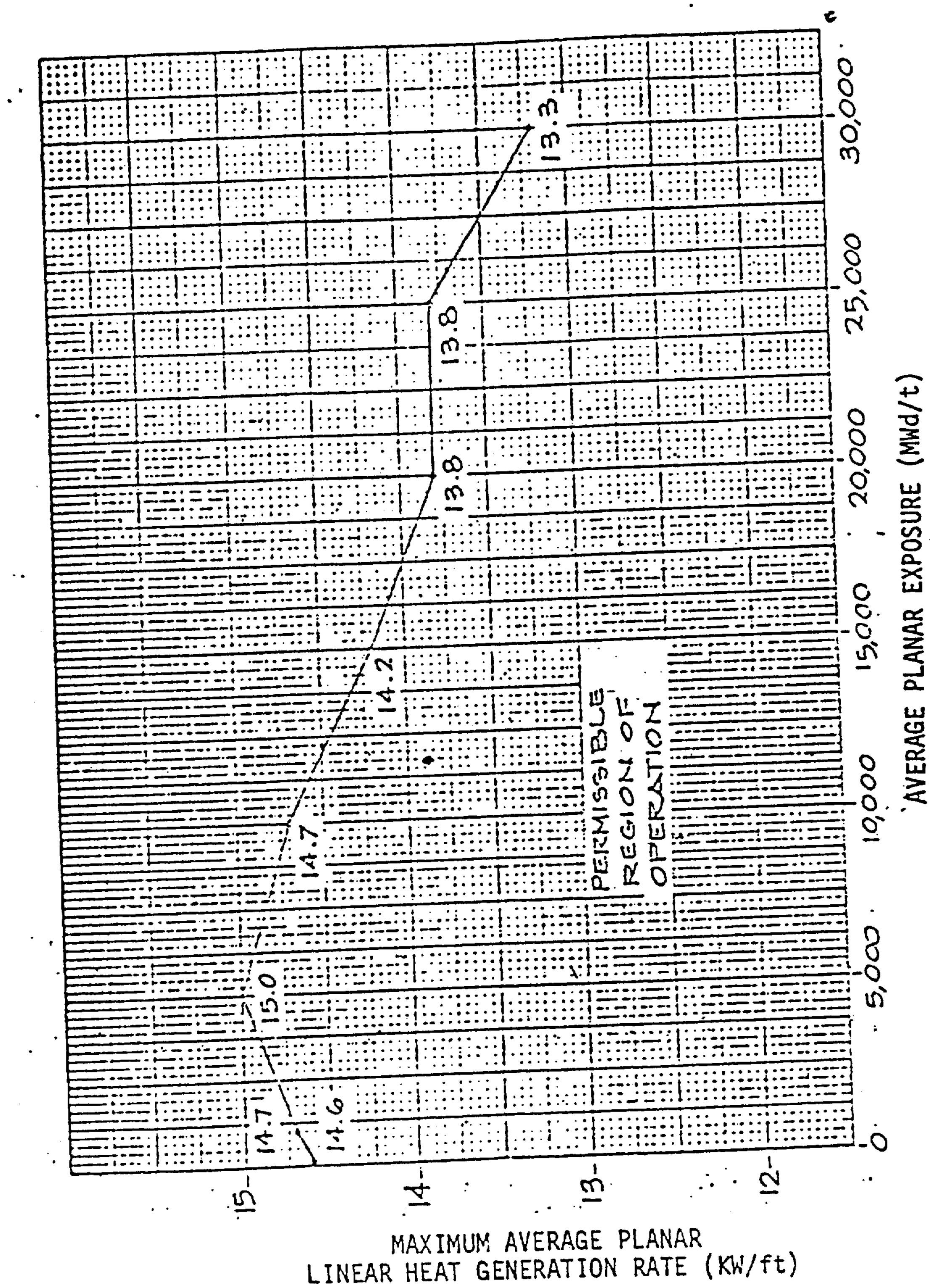
b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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



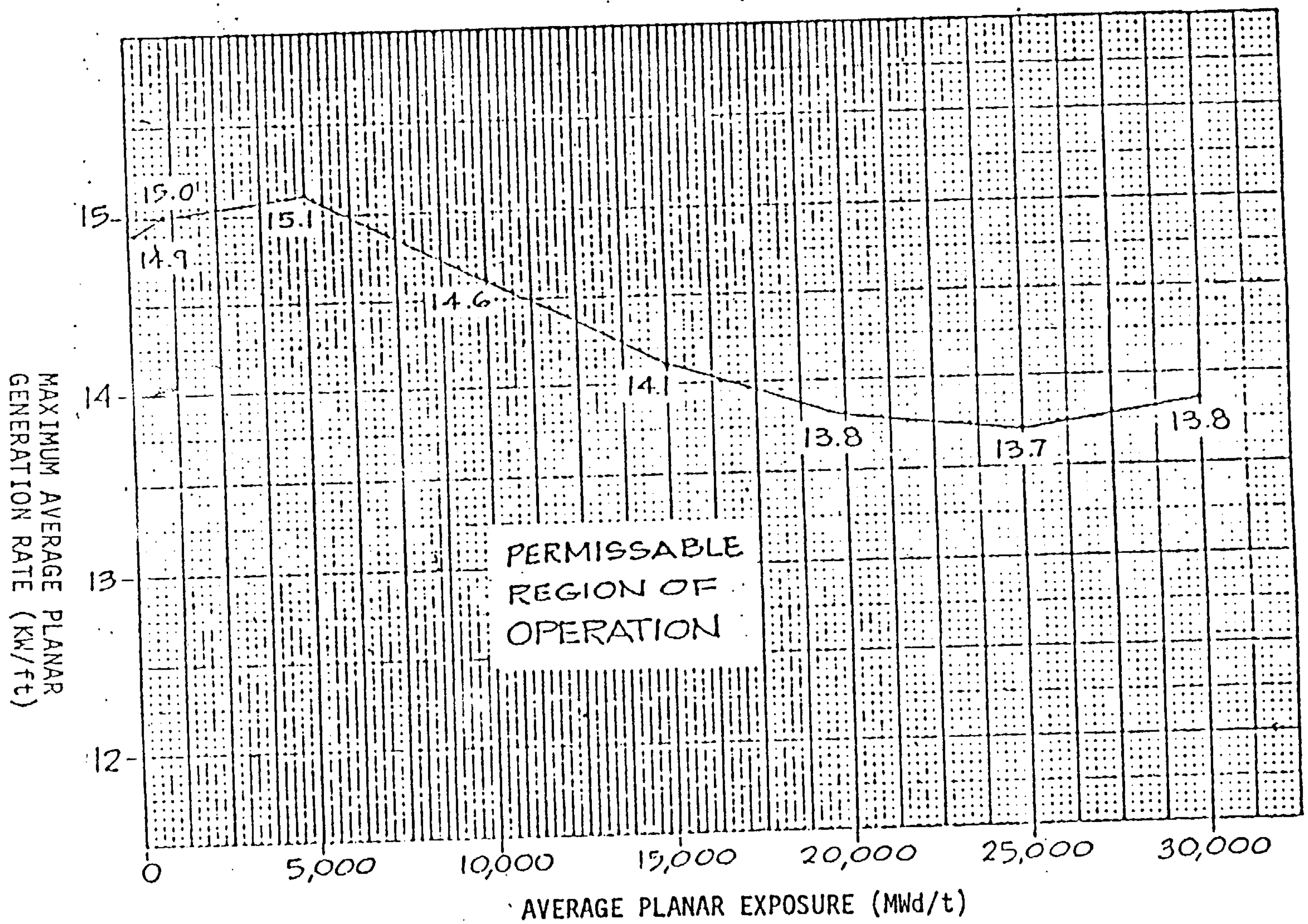
FUEL TYPE 1 & 2 (7x7)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-1



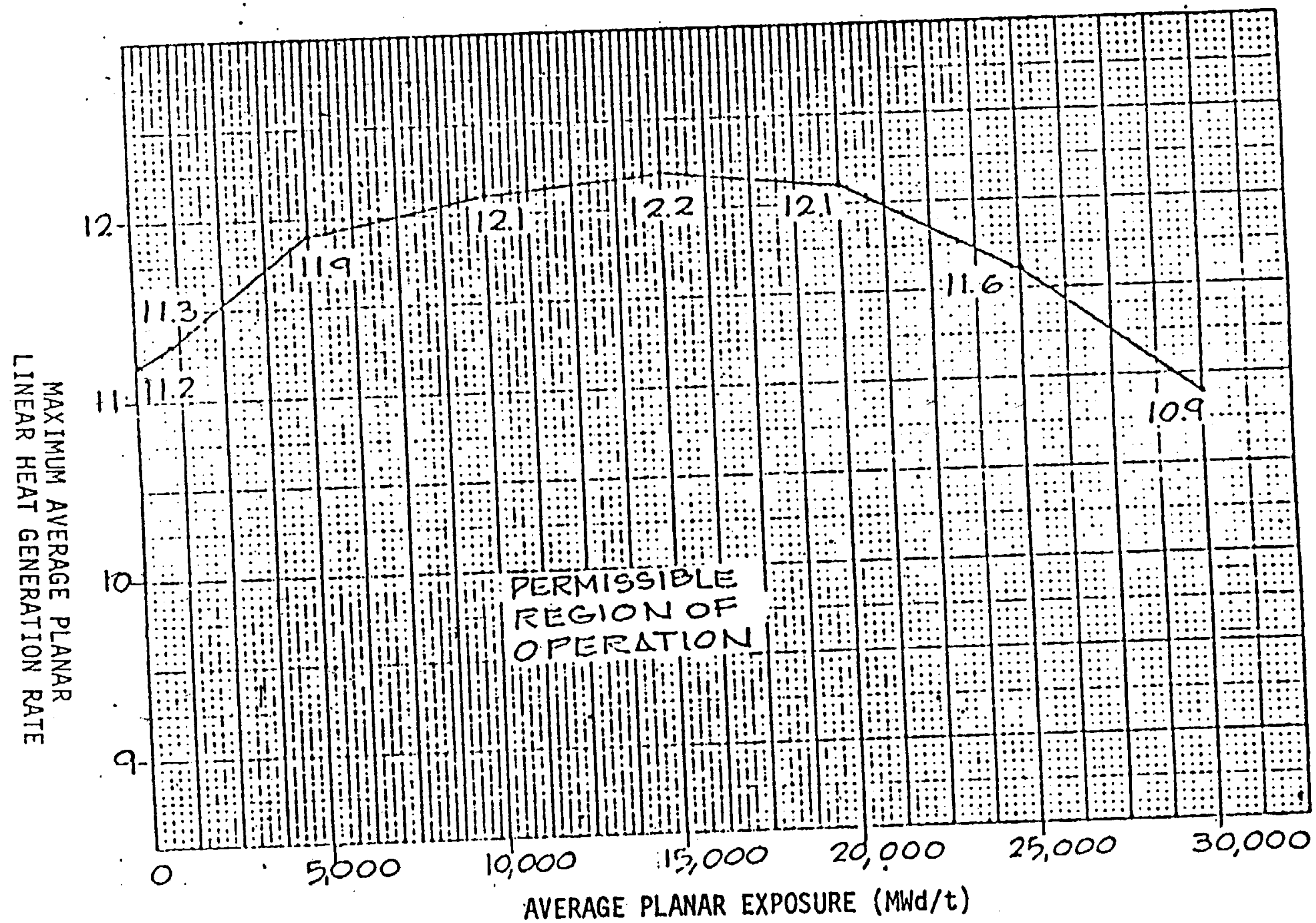
FUEL TYPE 3 (7x7)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-2



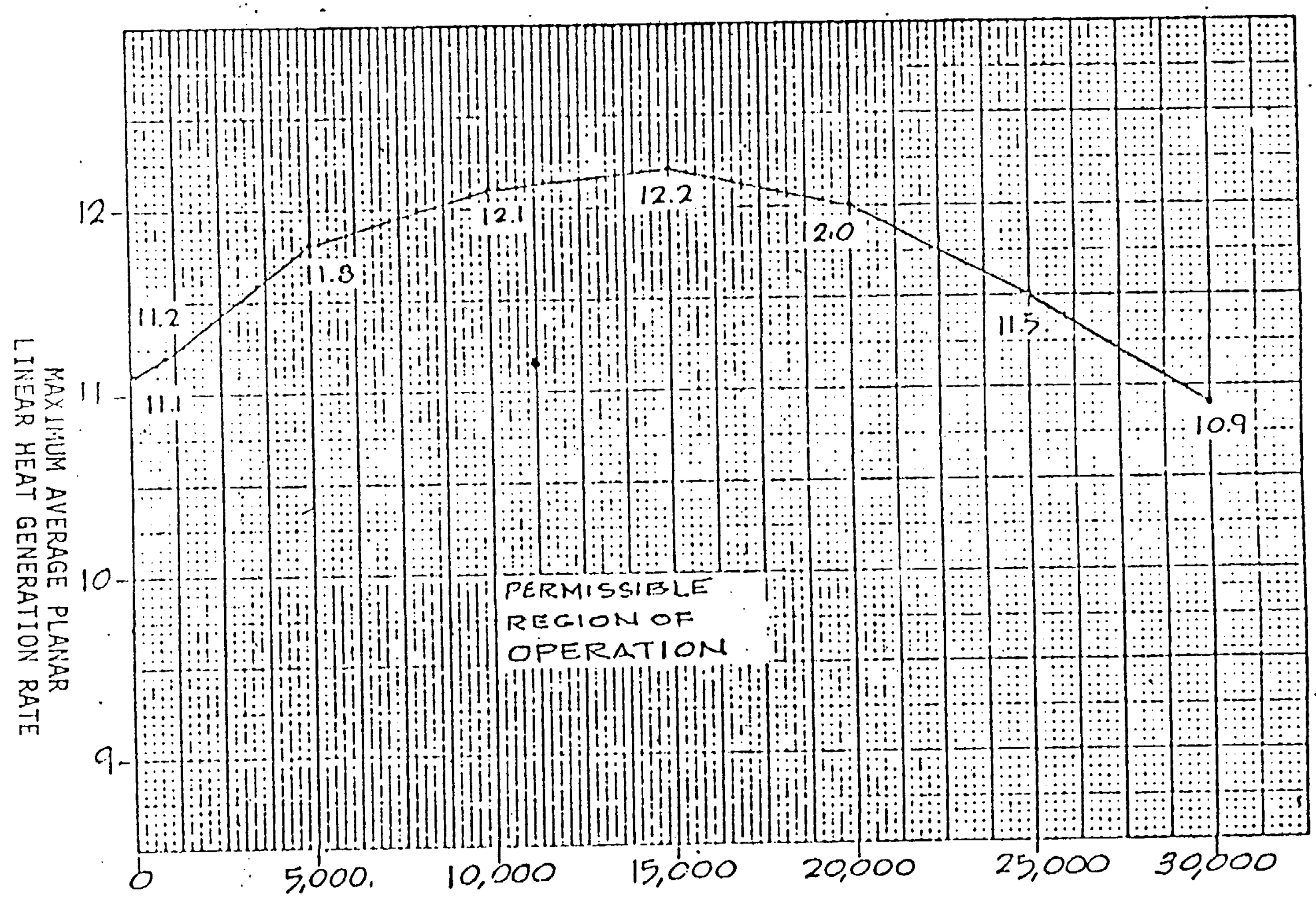
FUEL TYPE 7D230 (7x7)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-3



FUEL TYPE 8D274L (8x8)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

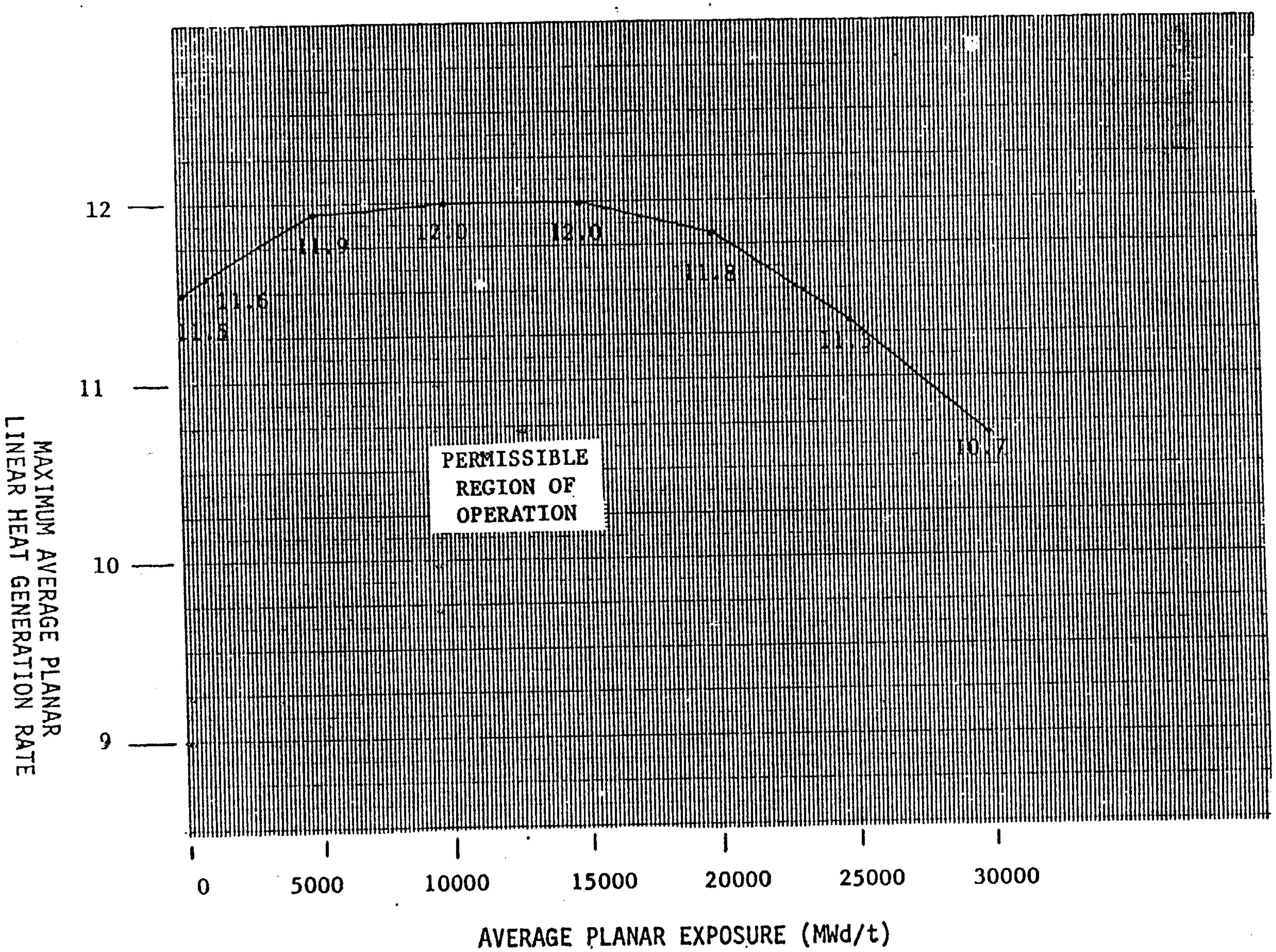
FIGURE 3.2.1-4



AVERAGE PLANAR EXPOSURE (MWd/t)

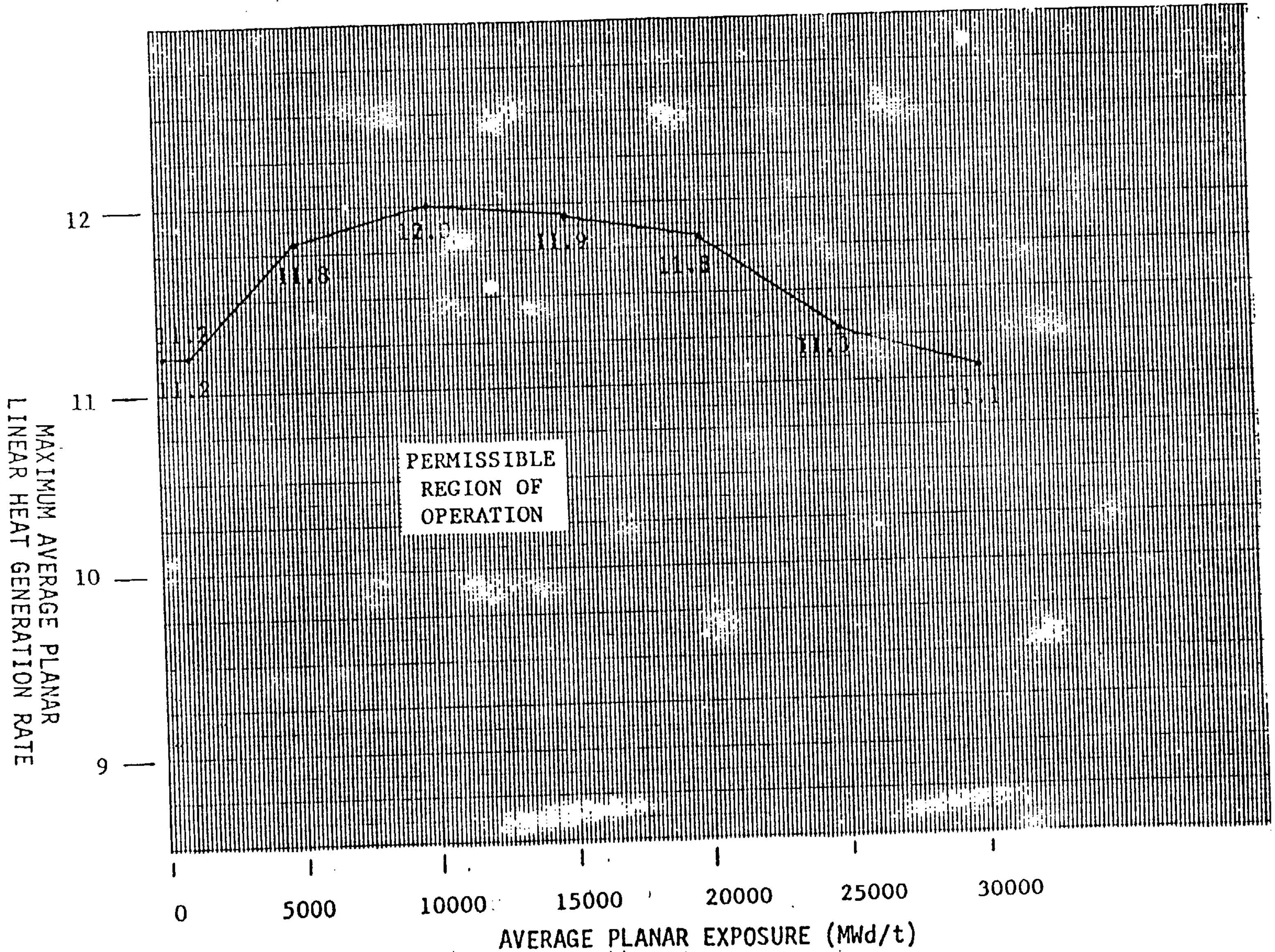
FUEL TYPE 8D274H (8x8)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-5



FUEL TYPE 8DRB265
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-6



FUEL TYPE 8DRB283
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 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 percent of rated flow,
T = Lowest value of the ratio of design TPF divided by the MTPF obtained for any class of fuel in the core ($T \leq 1.0$), and

Design TPF for: 8 x 8R fuel = 2.48
7 x 7 fuel = 2.60
8 x 8 fuel = 2.45

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. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, 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 times the K_f shown in Figure 3.2.3-1, for

a. Beginning-of-cycle (BOC) to end-of-cycle (EOC) minus 2000 MWD/t, with:

   1. MCPR for 7x7 fuel = 1.20,
   2. MCPR for 8x8 fuel = 1.21,
   3. MCPR for 8x8R fuel = 1.26.

b. EOC minus 2000 MWD/t to EOC, with:

   1. MCPR for 7x7 fuel = 1.21,
   2. MCPR for 8x8 and 8x8R fuel = 1.27.

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

ACTION:

With MCPR 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.

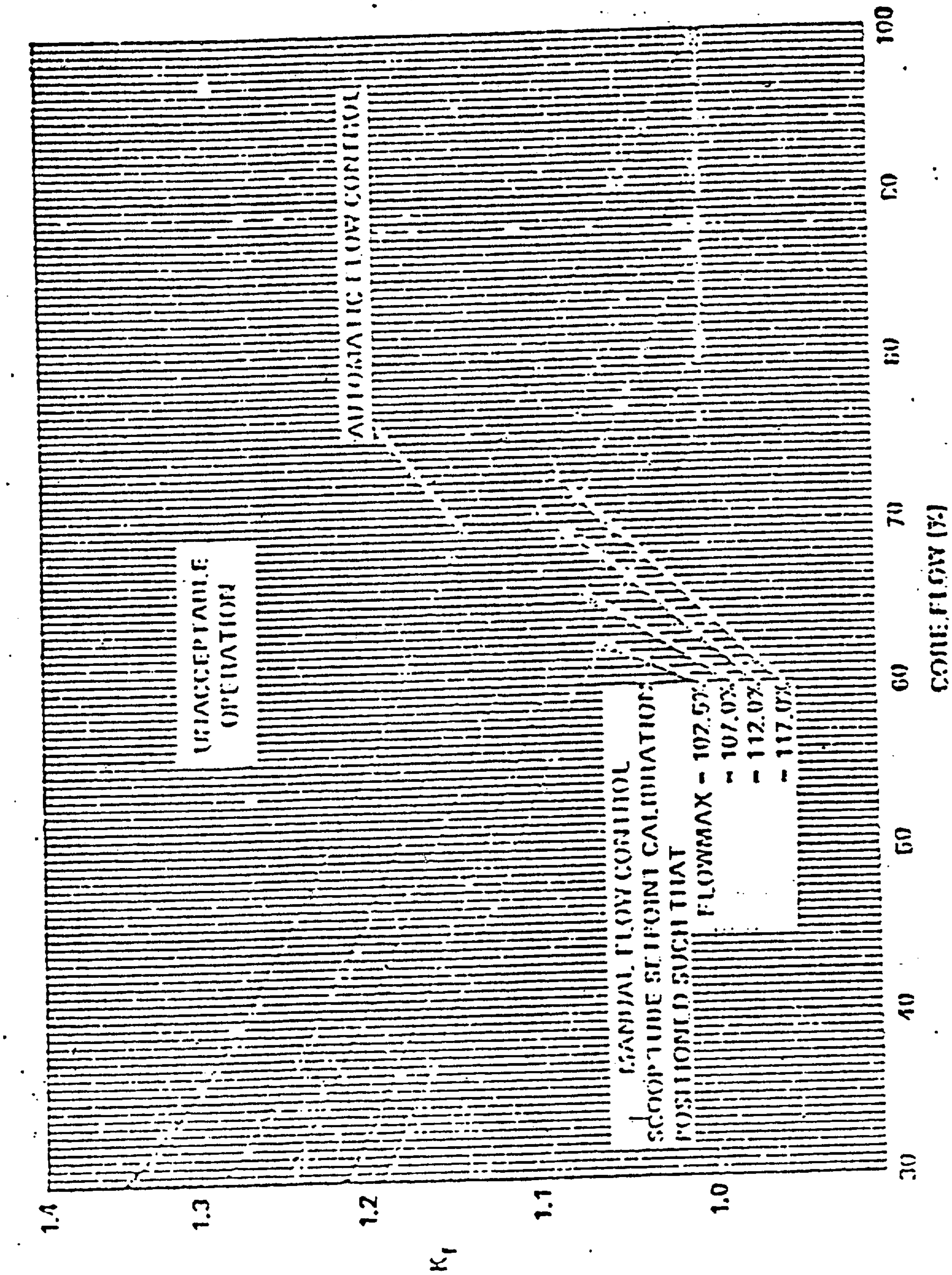
SURVEILLANCE REQUIREMENTS

4.2.3 MCPR 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. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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



K_f 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:

a. For 7 X 7 fuel assemblies, as a function of core height for any fuel rod in an assembly, the maximum allowable LHGR shown in Figure 3.2.4-1.

b. For 8 X 8 and 8 X 8R fuel assemblies, 13.4 kw/ft.

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

ACTION:

With the LHGR of any fuel rod exceeding the above limits, 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.

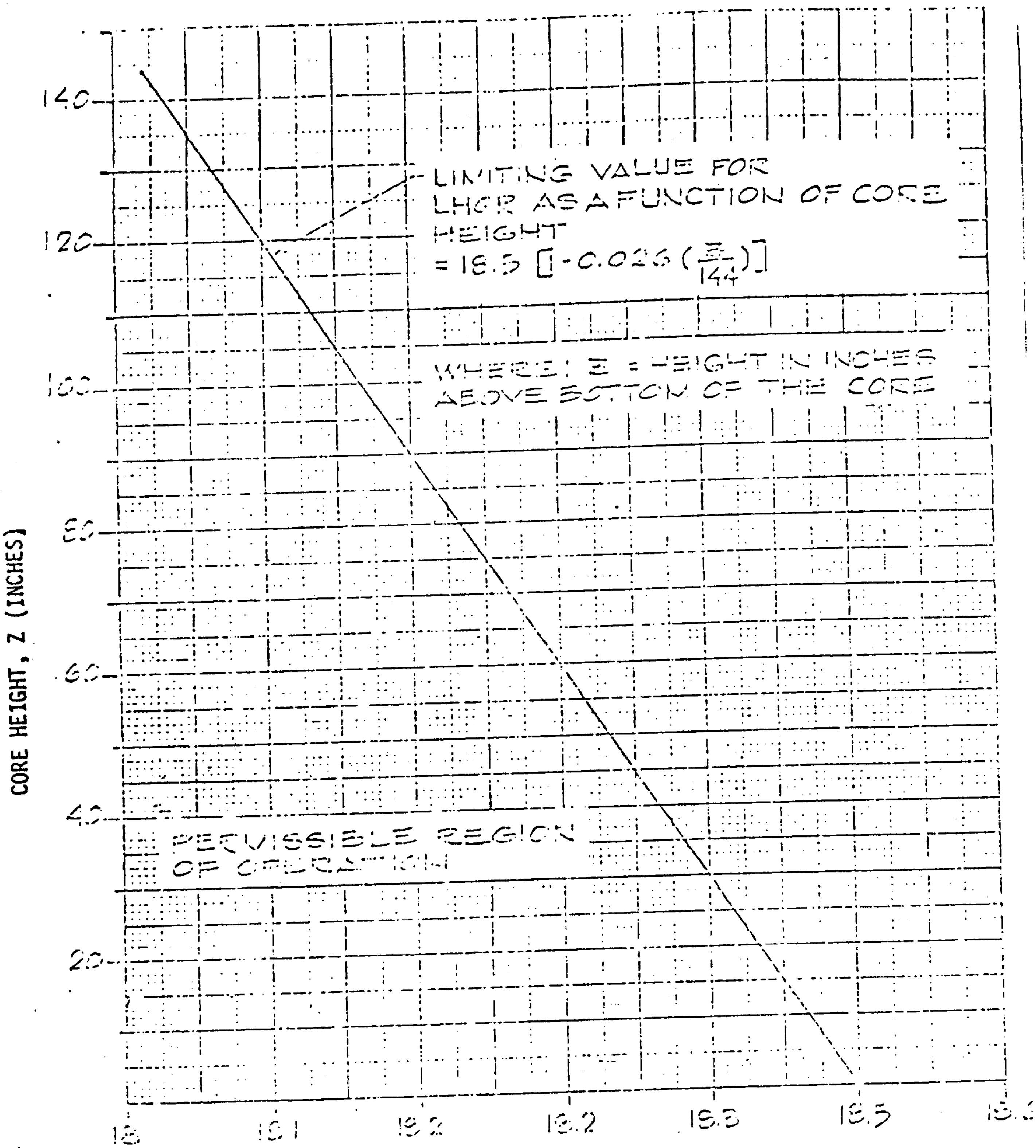
SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the applicable above limit:

a. At least once per 24 hours,

b. When THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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



LHGR (kw/ft)
LIMITING VALUE FOR LHGR
7 x 7 FUEL

FIGURE 3.2.4-1

3/4.3.4 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4 The control rod withdrawal block instrumentation shown in Table 3.3.4-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4-2.

APPLICABILITY: As shown in Table 3.3.4-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4-2, declare the channel inoperable until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, POWER OPERATION may continue provided that either:

   1. The inoperable channel(s) is restored to OPERABLE status within 24 hours, or

   2. The redundant trip system is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable channel is restored to OPERABLE status, and the inoperable channel is restored to OPERABLE status within 7 days, or

   3. For the Rod Block Monitor only, THERMAL POWER is limited such that MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of any single control rod that is capable of withdrawal.

   4. Otherwise, place at least one trip system in the tripped condition within the next hour.

c. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, place at least one trip system in the tripped condition within one hour.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.4 Each of the above required control rod withdrawal block instrumentation channels shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK, CHANNEL CALIBRATION and a CHANNEL FUNCTIONAL TEST during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.4-1.

TABLE 3.3.4-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OF OPERABLE CHANNELS PER TRIP SYSTEM[a]	APPLICABLE OPERATIONAL CONDITIONS
1. APRM (C51-APRM-CH.A,B,C,D,E,F)		
a. Upscale (Flow Biased)	2	1
b. Inoperative	2	1, 2, 5
c. Downscale	2	1
d. Upscale (Fixed)	2	2, 5
2. ROD BLOCK MONITOR (C51-RBM-CH.A,B)		
a. Upscale	1	1*
b. Inoperative	1	1*
c. Downscale	1	1*
3. SOURCE RANGE MONITORS (C51-SRM-K600A,B,C,D)		
a. Detector not full in[b]	1	2, 5
b. Upscale[c]	1	2, 5
c. Inoperative[c]	1	2, 5
d. Downscale[b]	1	2, 5
4. INTERMEDIATE RANGE MONITORS[d] (C51-IRM-K601A,B,C,D,E,F,G,H)		
a. Detector not full in[e]	2	2, 5
b. Upscale	2	2, 5
c. Inoperable	2	2, 5
d. Downscale[e]	2	2

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	$\leq (0.66W + 39\%)\ \frac{T^*}{MTPF}$	$\leq (0.66\ W + 39\%)\ \frac{T^*}{MTPF}$
b. Inoperative	NA	NA
c. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
3. SOURCE RANGE MONITORS (C51-SRM-K600A,B,C,D)		
a. Detector not full in	NA	NA
b. Upscale	$< 1 \times 10^5$ 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	$\leq$ 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

*T=2.60 for 7 x 7 fuel.
T=2.45 for 8 x 8 fuel.
T=2.48 for 8 x 8R fuel.

TABLE 4.3.4-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. APRM (C51-APRM-CH.A,B,C,D,E,F)				
a. Upscale (Flow Biased)	NA	S/U[c], M	R[b]	1
b. Inoperative	NA	S/U[c],Q	NA	1, 2, 5
c. Downscale	NA	S/U[c],M	NA	1
d. Upscale (Fixed)	NA	S/U[c],Q	R	2, 5
2. ROD BLOCK MONITOR (C51-RBM-CH.A,B)				
a. Upscale	NA	S/U[c],M	R	1*
b. Inoperative	NA	S/U[c],Q	NA	1*
c. Downscale	NA	S/U[c],M	R	1*
3. SOURCE RANGE MONITORS (C51-SRM-K600A,B,C,D)				
a. Detector not full in	NA	S/U[c],W	NA	2, 5
b. Upscale	NA	S/U[c],W	NA	2, 5
c. Inoperative	NA	S/U[c],W	NA	2, 5
d. Downscale	NA	S/U[c],W	NA	2, 5
4. INTERMEDIATE RANGE MONITORS (C51-IRM-K601A,B,C,D,E,F,G,H)				
a. Detector not full in	NA	S/U[c],W[d]	NA	2
	NA	W	NA	5
b. Upscale	NA	S/U[c],W[d]	NA	2
	NA	W	NA	5
c. Inoperative	NA	S/U[c],W[d]	NA	2
	NA	W	NA	5
d. Downscale	NA	S/U[c],W[d]	NA	2
	NA	W	NA	5

a. CHANNEL CALIBRATIONS are electronic.
b. This calibration shall consist of the adjustment of the APRM flow biased setpoint to conform to a calibrated flow signal.
c. Within 24 hours prior to startup, if not performed within the previous 7 days.
d. When changing from CONDITION 1 to CONDITION 2, perform the required surveill[illegible] within 12 hours after entering CONDITION 2.
* When THERMAL POWER is greater than the preset power level of the R[illegible] RSCS.

3/4.3.5 MONITORING INSTRUMENTATION

SEISMIC MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.1 The seismic monitoring instrumentation shown in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 31 days, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission, within the next 14 days outlining the cause of the malfunction and the plans for restoring the instruments to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.1.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.5.1-1.

4.3.5.1.2 Each of the above required seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. In lieu of any other report required by Specification 6.9.1, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

TABLE 3.3.5.7-1 (Continued)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE FLAME	HEAT	SMOKE
4. Service Water Building			
Zone 1 4'	0	0	6
Zone 2 20	0	0	5
5. AOG Building			
Zone 1 20'	1	0	0
Zone 2 20'	1	0	0
Zone 3 20'	1	5	1
Zone 4 37' - 49'	1	6	0

3/4.3.6 ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.1 The Anticipated Transient Without Scram recirculation pump trip (ATWS-RPT) system instrumentation trip systems shown in Table 3.3.6.1-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.1-2.

APPLICABILITY: CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6.1-2, declare the trip system inoperable until the trip system is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE trip systems per operating pump not satisfied for one Trip Function, restore the inoperable trip system to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.3.6.1.1 Each ATWS recirculation pump trip system instrumentation trip system shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.1.1-1.

4.3.6.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

TABLE 3.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE TRIP SYSTEMS PER OPERATING PUMP
1. Reactor Vessel Water Level - Low Low, Level 2 (B21-LIS-N024 A, B; B21-LIS-N025 A, B)	1
2. Reactor Vessel Pressure-Low (B21-PS-N045 A, B, C, D)	1

TABLE 3.3.6.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low low, Level 2 (B21-LIS-N024 A, B; B21-LIS-N025 A, B)	$\geq$ -38 inches	$\geq$ -38 inches
2. Reactor Vessel Pressure-Low (B21-PS-N045 A, B, C, D)	$\geq$1120 psig	$\geq$1120 psig

TABLE 4.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2 (B21-LIS-N024 A, B; B21-LIS-N025 A, B)	S	M	R
2. Reactor Vessel Pressure - Low (B21-PS-N045 A, B, C, D)	NA	M	R

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold xenon-free condition and shall show the core to be subcritical by at least R + 0.38% ΔK. The value of R in units of %ΔK is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle. Satisfaction of this limitation can be best demonstrated at the time of fuel loading but the margin must be determined any time a control rod is incapable of insertion.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

3/4.1.3 CONTROL RODS

The specifications of this section ensure that 1) the minimum SHUTDOWN MARGIN is maintained, 2) the control rod insertion times are consistent with those used in the accident analysis, and 3) the

CONTROL RODS (Continued)

potential effects of the rod ejection accident are limited. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the non-fully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than 1.07 during the limiting power transient analyzed in Section 14.3 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MPCR remains greater than 1.07. The occurrence of scram times longer than those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion

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 tempeature 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, 3.2.1-6 and 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 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

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	8 x 8	13.4	1.4	1.20
	7 x 7	18.5	1.5	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.

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 2.60 for 7 x 7 fuel, 2.45 for 8 x 8 fuel and 2.48 for 8 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 2.60 for 7 x 7 fuel, 2.45 for 8 x 8 fuel and 2.48 for 8 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 bypass. This transient yields the largest Δ MCPR. When added to the Safety Limit MCPR of 1.07 the required minimum operating limit MCPR of Specification 3.2.3 is obtained. Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE 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.

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.

MINIMUM CRITICAL POWER RATIO (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.2.3-1 are conservative for the General Electric Plant operation with 8 x 8 and 8 x 8R fuel assemblies because the operating limit MCPR's of Specification 3.2.3 are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f. The k_f curves are conservative for 7 x 7 fuel whenever the operating limit MCPR is greater than 1.23 as documented in Appendix C of NEDE 24011-P-A. A correction to the K_f curves is, therefore, necessary whenever the MCPR for the 7 x 7 fuel is equal to or less than 1.23 in order to ensure that the fuel cladding integrity Safety Limit is not violated. This correction is made by using a scoop tube set point of 102.5%. The MCPR for 7 x 7 fuel is then the product of the value given in Specification 3.2.3 and the K_f curve based on 112% as shown in Figure 3.2.3-1. Whenever the MCPR for the 7 x 7 fuel is greater than 1.23, this correction is not applied.

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with mimimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape, regardless of magnitude that could place operation at a thermal limit.

3.2.4 LINEAR HEAT GENERATION RATE

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

BASES

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566, January, 1976.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

3. Letter from J. A. Jones, Carolina Power and Light Company to B. C. Rusche, NRC transmitting Amendment 31 to the Brunswick Unit 1 Docket No. 50-325, dated November 26, 1975.

4. General Electric BWR Generic Reload Application for 8 x 8 Fuel, NEDO-20360, Revision 1, November 1974.

5. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

6. Letter from J. A. Jones, Carolina Power and Light Company, to B. C. Rusche, NRC dated May 7, 1976.

MONITORING INSTRUMENTATION (Continued)

3/4.3.5.2 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of CFR 50.

3/4.3.5.3 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97 "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975.

3/4.3.5.4 SOURCE RANGE MONITORS

The source range monitors provide the operator with information on the status of the neutron level in the core at very low power levels during startup. At these power levels reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adequate information is available without the SRM's and they can be retracted.

3/4.3.5.5 CHLORINE DETECTION SYSTEM

The OPERABILITY of the chlorine detection systems ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine the control room emergency ventilation system will automatically isolate the control room and initiate operation in the recirculation mode to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

MONITORING INSTRUMENTATION (Continued)

3/4.3.5.6 CHLORIDE INTRUSION MONITORS

The chloride intrusion monitors provide adequate warning of any leakage in the condenser or hotwell so that actions can be taken to mitigate the consequences of such intrusion in the reactor coolant system. With only a minimum number of instruments available increased sampling frequency provides adequate information for the same purpose.

3/4.3.5.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.6 ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

The ATWS recirculation pump trip system has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within an envelope of study events given in General Electric Company Topical Report NEDO-10349, dated March, 1971.

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1, based on the information given in Section 2.2 of the FSAR.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The PRIMARY CONTAINMENT is a steel lined reinforced concrete structure composed of a series of vertical right cylinders and truncated cones which form a drywell. This drywell is attached to a suppression chamber through a series of vents. The suppression chamber is a concrete steel lined pressure vessel in the shape of a torus. The primary containment has a minimum free air volume of (288,000) cubic feet.

DESIGN TEMPERATURE AND PRESSURE

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

a. Maximum internal pressure 62 psig.

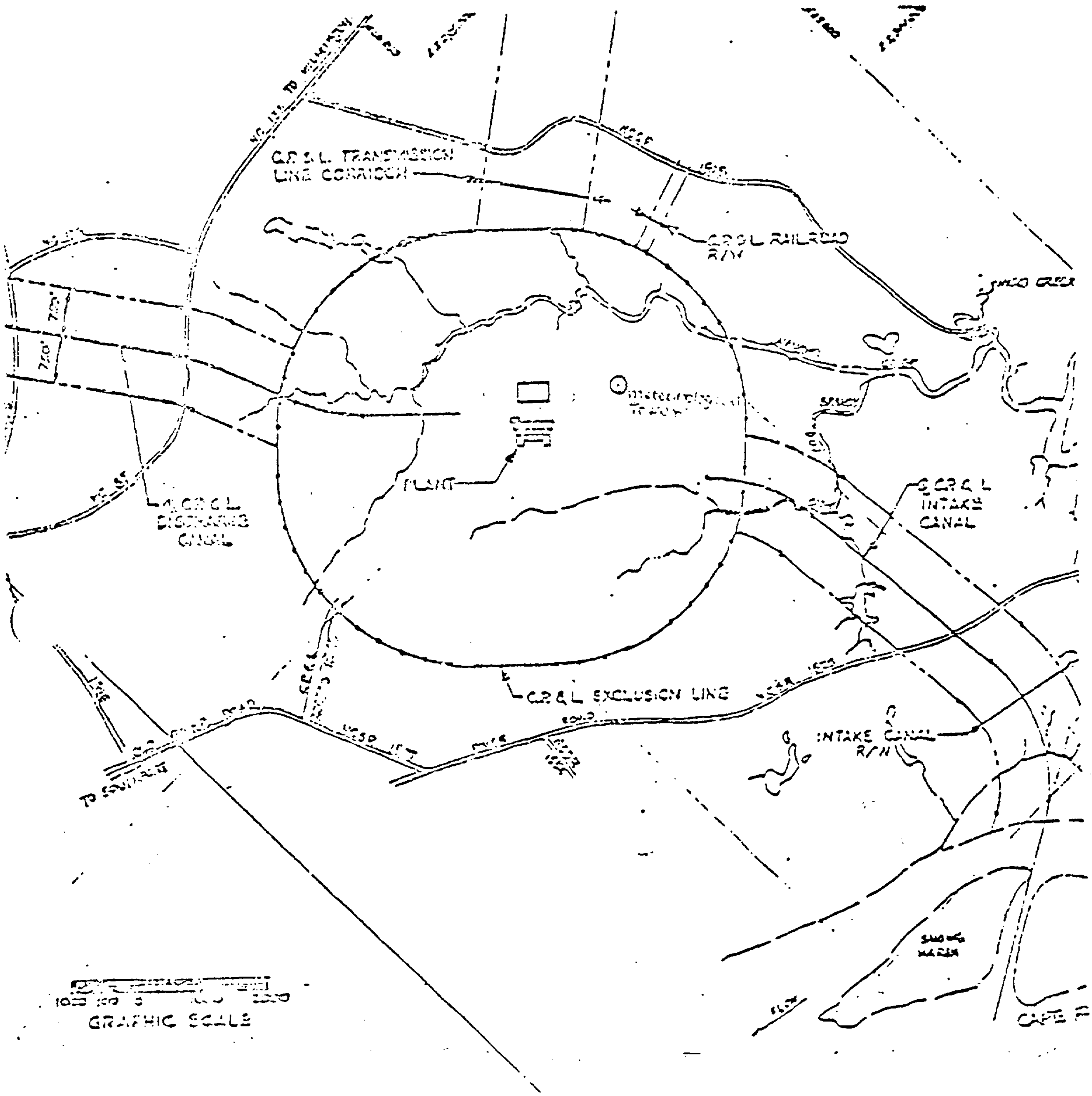
b. Maximum internal temperature: drywell 300°F.
suppression chamber 200°F.

c. Maximum external pressure 2 psig.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 560 fuel assemblies with each 7 x 7 fuel assembly containing 49 fuel rods, each 8 x 8 fuel assembly containing 63 fuel rods; and each 8 x 8R fuel assembly containing 62 fuel rods. All fuel rods shall be clad with Zircaloy 2. The nominal active fuel length of each fuel rod shall be 144 inches for 7 x 7 fuel assemblies, 146 inches for 8 x 8 fuel assemblies, and 150 inches for 8 x 8R fuel assemblies. Each fuel rod shall contain a maximum total weight of 4430 grams of UO_2.



EXCLUSION AREA

Figure 5.1.1-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER AND LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-324

1.0 . Introduction

By letter dated February 2, 1979, as supplemented March 16, 21 and 27, April 13, April 27, and May 1, 1979, Carolina Power and Light Company (the licensee) requested amendments to Facility Operating License No. DPR-62. The amendments would modify the Technical Specifications for the Brunswick Steam Electric Plant, Unit No. 2 (the facility) to establish revised safety and operating limits for operation in Cycle 3 with 7x7, 8x8, and 8x8R fuel. The February 2, 1979 submittal requested credit for the end-of-cycle recirculation pump trip (EOC RPT) feature which was installed in BSEP Unit No. 2 during the refueling outage. The April 13, 1979 submittal provided a non-RPT analysis for Unit No. 2 Cycle 3. This analysis was performed to generate fallback operating MCPR limits if the EOC RPT becomes inoperable during Cycle 3.

As a result of the licensee's proposal and our review we have some reservations about the design of the EOC RPT system. Therefore, as agreed to with your staff, we have not included credit for the EOC RPT system in the operating limit minimum critical power ratios. As a result, modifications to the licensee's proposed Technical Specifications were necessary. These modifications were discussed with and agreed to by the licensee.

2.0 Discussion

The Carolina Power and Light Company has proposed changes to the Technical Specifications of the Brunswick Unit No. 2 Nuclear Power Plant (BSEP 2). The proposed changes relate to the replacement of 132 fuel assemblies constituting refueling of the core for third cycle operation at power levels up to 2436 Mwt (100% power).

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In support of the reload application, the licensee has provided the GE BWR Reload 2 Licensing submittal for BSEP 2 (References 1, 2), information on the BSEP 2 Loss-of-Coolant Accident (LOCA) analysis (References 1 and 3), responses to NRC requests for additional information (Reference 12), and BSEP 2 Physics Startup Tests (Reference 5).

This reload involves loading of General Electric Company Retrofit (8x8R) fuel. The description of the nuclear and mechanical design of the (8x8R) fuel and the (8x8) fuel is contained in GE's licensing topical report for BWR reloads (Reference 6). Reference 6 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing (7x7), (8x8) and (8x8R) fuel.

Values for each plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other various design parameters are provided in Reference 6.

Additional plant and cycle dependent information are provided in the reload application (Reference 1) which closely follows the outline of Appendix A of Reference 6.

Reference 8 describes the staff's review, approval, and conditions of approval for the plant-specific data addressed in Reference 6. The above mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

Our Safety Evaluation (Reference 8) of the GE generic reload licensing topical report concluded that the nuclear and mechanical design of the (8x8R) fuel, and GEs analytical methods for nuclear, thermal-hydraulic, and transient and accident calculations as applied to mixed cores containing (7x7), (8x8) and (8x8R) fuel are acceptable. Approval of the nuclear and mechanical design of (8x8) fuel was determined based on information in Reference 7 and expressed in the staff's status report (Reference 9) on that document.

Because of our review of a large number of generic considerations related to use of (8x8R) fuel in mixed loadings with (8x8) and (7x7) fuel, and on the basis of the evaluations which have been presented in Reference 8, only a limited number of additional areas of review have been included in this safety evaluation report. For evaluations of areas not specifically addressed in this safety evaluation report, the reader is referred to Reference 8.

During the current outage CP&L has modified Unit No. 2 to provide automatic trip of both recirculation pumps after turbine trip or generator load rejection. The purpose of this trip is to reduce the reactor pressure and peak heat flux resulting from these transients coincident with a failure of the bypass system. Our safety evaluation (Reference 8) did not include an evaluation of the prompt recirculation trip (RPT) proposed by BSEP 2.

Because several issues remain unresolved regarding the implementation of the proposed EOC RPT system at BSEP 2, CP&L requested approval with no credit for the EOC RPT thermal margin improvements (Reference 12). Our review and approval of the BSEP 2 operating limits are discussed in Section 3.2.2.

3.0 Evaluation

3.1 Nuclear Characteristics

For Cycle 3 operation of BSEP2, 64 (8x8R) fuel bundles of type 8DR B 265H and 68 (8x8R) bundles of type 8DR B 283 will be loaded into the core (Reference 1). The remainder of the 560 fuel bundles in the core will be fuel used during the previous cycle.

The fresh fuel will be loaded in a core pattern as shown in Figure 1 of Reference 1, which is acceptable.

Based on the data presented in sections 4 and 5 of Reference 1, both the control rod system and the standby liquid control system will have acceptable shutdown capability during Cycle 3.

3.2 Thermal Hydraulics

3.2.1 Fuel Cladding Integrity Safety Limit

As stated in Reference 6, the minimum critical power ratio (MCPR) which may be allowed to result from core-wide or localized transients is 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid transition boiling.

The safety limit MCPR for BSEP2 is being raised to 1.07 because the distribution of fuel rod power with the (8x8R) fuel bundles is flatter than that of the (8x8) fuel. The reason for the flatter power distribution is the presence of two rather than one water rods in (8x8R) fuel. The issue has been addressed in Reference 8 and the 1.07 limit has been found acceptable for BWRs with uncertainties in flux monitoring and operational parameters no greater than those listed in Table 5-1 of Reference 6, for which the CPR distribution is within the bounds of Figures 5.2 and 5.2a of Reference 6. It has been shown in Reference 1 that these conditions are met for BSEP2 Cycle 3.

In addition to the 1.07 MCPR safety limit discussed above, the (8x8) and (8x8R) fuel must be maintained within the 17.5 KW/ft exposure-dependent Linear Heat Generation Rate (LHGR) safety limit. Maximum LHGR conditions can occur during abnormal operational conditions which affect the fuel locally, e.g., Rod Withdrawal Error and the Fuel Loading Error. In this regard, the staff requires that the calculated maximum transient LHGR for the 8x8 and 8x8R fuel be augmented by a fuel densification power spike allowance. As stated in Reference 11 since implementation of this requirement for BSEP2 meets the exposure-dependent safety limit for the 8x8 and 8x8R fuel, the staff finds it acceptable that the 8x8 and 8x8R fuel densification power spike penalty be deleted from the BSEP2 Technical Specifications.

Because the (7x7) fuel was designed before fuel densification and its effects were known, the newly implemented and revised GE analytical procedures to mechanistically account for densification power spiking do not apply to the (7x7) fuel. Therefore, the power spiking penalty, as included in the Technical Specifications, shall continue to be used for the (7x7) fuel.

3.2.2 Operating Limit MCPR

Various transients could reduce the CPR below the intended operating limit MCPR during Cycle 3 operation. The most limiting of these operational transients and also the potential fuel loading errors have been analyzed by the licensee to determine which event could induce the largest reduction in the critical power ratio (ΔCPR).

The transients evaluated were the generator load rejection without bypass, turbine trip without bypass, feedwater controller failure at maximum demand, inadvertent HPCI pump startup, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Tables 6 and 7 of References 1 and 2 were assumed.

In the analyses of these reactor transients, the licensee submitted results based on transients that include the prompt RPT feature, and transients which take no credit for the RPT being operable.

As stated in Section 1.0, several issues remain unresolved relative to the implementation of the proposed EOC RPT system. Therefore, we cannot give credit for the reductions in operating limits (MCPR) afforded by the EOC RPT feature. Since the turbine trip and generator load rejection transients without bypass represent the limiting transients only near the EOC, with all control rods withdrawn, we can approve the earlier part of Cycle 3 (BOC to EOC - 2000 MWD/t) based on other transients which are most limiting over this interval. For the later part of Cycle 3, (EOC - 2000 MWD/t to EOC), the operating limit MCPR will be based on the most limiting of these two transients without benefit of the EOC RPT reductions in thermal margin (ΔCPR).

However, even though we cannot give credit at this time for the EOC RPT installation at BSEP 2, we do believe it prudent that CP&L perform functional testing of the EOC RPT operational aspects, e.g., flow coastdown rate and time response measurements during their startup test program. These tests should give the necessary information to provide assurance that the EOC RPT system will perform within the bounds of the analysis. In addition, even though we have not given credit for this feature in this safety analysis for the reasons previously stated, we recognize the potential benefits afforded by the immediate reduction in core flow with increased core voiding and the resultant negative reactivity. Therefore, until we can approve the implementation of the EOC RPT system at BSEP 2, operation of the "as built" EOC RPT should provide an extra margin of conservatism in BSEP 2 operating limits. We thus have no objection to the use of this system during Cycle 3, provided the licensee performs the appropriate 10 CFR 50.59 evaluation.

As shown below, addition of the highest ΔCPR resulting from the most severe transient during the specified exposure interval to the safety limit (1.07) gives the appropriate operating limit MCPR for each fuel type. This sum will assure that the safety limit is not violated during Cycle 3 operation at BSEP 2.

Limiting Transient	Exposure Interval	ΔCPR (7x7)/(8x8)/(8x8R)	MCPR Operating Limit w/o RPT
Rod with-drawal error	BOC to EOC-2 GWD/t	.13/**/.19	1.20*/**/1.26
Inadvertent HPCI Pump start	BOC to EOC-2 GWD/t	**/.14/**	**/1.21/**
Generator Load rejection w/o bypass	EOC-2 GWD/t to EOC	.14/.20/.20	1.21*/1.27/1.27

*For the 7x7 fuel, the 102.5% core flow K_f curve is nonconservative (Reference 6) with operating limits <1.23. Therefore, at reduced flow conditions, the K_f factor for the 7x7 fuel assemblies will be based on the 112% flow curve of Figure 3.2.2-1 of the Technical Specifications rather than the actual setpoint of 102.5%.

**Not limiting

We have determined that the operating limit MCPRs listed above are acceptable for Cycle 3 operation at the BSEP 2 plant.

3.3 Overpressure Analaysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 8. As specified in Reference 8, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the overpressure design limit (1375 psi) to allow for the failure of at least one valve. Therefore the limiting overpressure event as analyzed by the licensee is acceptable.

3.4 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis (Reference 1) show that the channel hydrodynamic and reactor core decay ratios at the Natural Circulation - 105% Rod Line intersection (which is the least stable physically attainable point of operation) are below the 1.0 stability limit.

Because operation in the natural circulation mode is restricted by Technical Specifications, there will be added margin to the stability limit. We find this is acceptable.

3.5 Accident Analysis

3.5.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License, implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading..."the licensee shall submit a reevaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation assumptions.

The licensee has reevaluated the adequacy of ECCS performance in connection with the new reload fuel design, using methods previously approved by the staff. The results of these analyses are given in References 1, 2, and 3.

We have reviewed the information submitted by the licensee and conclude that all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 will be met when the reactor is operated in accordance with the MAPLHGR versus Average Planar Exposure values given in Figures 3.2.1-1, 2, 3, 4, 5, 6, and 7 of the Technical Specifications.

3.5.2 Control Rod Drop Accident

For BSEP2, Cycle 3, the accident reactivity insertion curves satisfy the requirements for the bounding analyses described in Reference 5. Therefore, the peak fuel enthalpy for this event would be less than 280 calories/gram, which is acceptable.

3.5.3 Fuel Loading Error

Potential fuel loading errors involving misoriented bundles have been explicitly included in the calculation of the operating limit MCPR. Potential errors involving bundles loaded into incorrect positions have also been analyzed by a method which considers the initial MCPR of each bundle in the core, and the resultant MCPR was shown to be greater than 1.07. The GE method for analysis of misoriented and misloaded bundles has been reviewed and approved by the staff (Reference 10).

The analyses which have been performed for potential fuel loading errors for BSEP2, Cycle 3, are acceptable for assuring that CPRs will not be below the safety limit MCPR of 1.07.

4.0 Physics Startup Testing

The safety analysis for the upcoming cycle is based upon a specifically designed core configuration. We have assumed that, after reloading, the actual core configuration will conform to the designed configuration. A startup test program can provide the assurance that the core conforms to the design. We require that a startup test program be performed and the minimum recommended tests are:

1. Visual inspection of the core using a photographic or videotape record.

2. A check of core power symmetry by checking for mismatches between symmetric detectors.

3. Withdrawal and insertion of each control rod to check for criticality and mobility.

4. Comparison of predicted and measured critical insequence rod pattern for nonvoided conditions.

We find the startup test program, (Reference 5), submitted by CP&L acceptable for Cycle 3 operation.

In the future, as a result of our ongoing generic review of BWR startup test, we anticipate requiring a description of each test sufficient to show how it provides assurance that the core conforms to the design. The description is anticipated to include both the acceptance criteria and the actions to be taken in case the acceptance criteria are not obtained.

In addition to the requirements, above, we request that a brief written report of the startup tests be submitted to the NRC within 45 days of the completion of the tests.

Environmental Considerations

We have determined that this 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 this 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 this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significnat decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: May 2, 1979

References

1. "Supplemental Reload Licensing Submittal" for Brunswick Unit 2 Reload 2, NEDO-24179-1, March 1979.

2. "Supplemental Reload Licensing Submittal NEDO-24182, March 1979.

3. Lead Plant LOCA Analysis: James A. FitzPatrick Nuclear Power Plant, July 1977 (NEDO-21662).

4. Letter, E. E. Utley (CPL) to T. A. Ippolito, (NRC), dated March 27, 1979 requesting deletion of Power Spiking Penalty.

5. Letter, E. E. Utley (CPL) to T. A. Ippolito, (NRC), dated March 16, 1979 transmitting Physics Startup Test Program.

6. General Electirc Boiling Water Reactor Generic Reload Fuel Application, NEDO-24011-P, May 1977.

7. General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel, NEDO-20360, Rev. 1, Supplement 4, April 1, 1976.

8. Safety Evaluation of the GE Generic Reload Fuel Application (NEDE-24011-P), April 1978.

9. Status Report on the Licensing Topical Report "General Electric Boiling Water Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, April 1975.

10. Safety Evaluation of New GE Fuel Loading Error Methods, April 1978.

11. Letter, D. G. Eisenhut (NRC), to R. Gridley (GE), dated June 9, 1978, transmitting: Safety Evaluation of the General Electric Methods for the Consideration of Power Spiking due to Densification Effects in BWR 8x8 Fuel Design and Performance.

12. Letter, E. E. Utley (CP&L) to T. A. Ippolito (NRC), dated May 1, 1979, requesting operating limits with no credit for the EOC recirculation pump trip.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-325 AND 50-324

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 24 and 48 to Facility Operating License Nos. DPR-71 and DPR-62 issued to Carolina Power & Light Company (the licensee) which revised the Technical Specifications for operation of the Brunswick Steam Electric Plant, Units Nos. 1 and 2 (the facility), located in Brunswick County, North Carolina. The amendments are effective as of the date of issuance.

The amendments for BSEP, Units 1 and 2 provide Technical Specifications for the protective instrumentation associated with the Anticipated Transients Without Scram Recirculation Pump Trip. These specifications were inadvertently omitted when Amendment No. 12 to DPR-71 and Amendment No. 39 to DPR-62 were issued on November 23, 1977.

The amendment for BSEP Unit 2 also changes the Technical Specifications to establish revised safety and operating limits for operation in Cycle 3 with 7x7, 8x8, and 8x8R fuel.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license

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amendments. Prior public notice of the amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated February 2, 1979, as supplemented March 16, 21 and 27, April 13 and 27, and May 1, 1979, (2) Amendment Nos. 24 and 48 to Licenses Nos. DPR-71 and DPR-62, and (3) the Commission's related Safety Evaluation. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Southport-Brunswick County Library, 109 West Moore Street, Southport, North Carolina 28461. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 2nd day of May 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors