February 6, 1989

Docket No. 50-325

DISTRIBUTION
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Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

SUBJECT: ISSUANCE OF AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1, REGARDING CYCLE 7 RELOAD (TAC NO. 69200)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 124 to Facility Operating License No. DPR-71 for the Brunswick Steam Electric Plant, Unit 1. The amendment consists of changes to the Technical Specifications in response to your reload submittal dated August 1, 1988. Environmental related information on extended fuel irradiation was also previously provided by letter dated September 25, 1987.

The amendment changes the Technical Specifications to: (1) revise the minimum critical power ratio (MCPR) safety limit, (2) modify operating limits for average power range monitor (APRM) setpoints, MCPR values, maximum average planar linear heat generation rate (MAPLHGR) values and linear heat generation rate (LHGR) requirements) for Cycle 7, (3) revise the values of mu and sigma found in Specification 3.2.3.2 to conform to the advanced GEMINI/ODYN analysis methods and add a reference to Notch 36 for Specification 3.2.3.2, (4) redefine Critical Power Ratio and Physics Tests, (5) permit fuel burnup not to exceed 60,000 MWD/MT and (6) change the bases statements accordingly to reflect the above described changes.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly <u>Federal Register</u> Notice.

Sincerely,

/S/

Edmond G. Tourigny, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000325
P PDC

Enclosures:
1. Amendment No. 124 to License No. DPR-71
2. Safety Evaluation

cc w/enclosures:
See next page

DFOI
1/1

CP add.

[BSEP1 AMEND 69200]

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:
NAME	:PAnderson	:ETourigny	:EReeves	:	:	:
DATE	: 1/25/89	: 2/6/89	: 02/06/89	:	:	:

Mr. E. E. Utley
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

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AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1

Docket File
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cc: Licensee/Applicant Service List

DF01
1/1

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124
License No. DPR-71

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated August 1, 1988 and associated letter dated September 25, 1987, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. DPR-71 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.124 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

EGA for

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 6, 1989

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR			
NAME	PAnderson	ELourigny		~~EAdensam~~ E. REEVES			
DATE	1/12/89	1/12/89	1/ /89	02/06/89	1/18/89		

OFFICIAL RECORD COPY

ATTACHMENT TO LICENSE AMENDMENT NO. 124

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

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

Remove Pages	Insert Pages
III	III
IV	IV
1-2	1-2
1-5	1-5
2-1	2-1
B2-1	B2-1
B2-2	B2-2
B2-3	B2-3
B2-4	B2-4
B2-5	B2-5
B2-6	B2-6
B2-7	B2-7
B2-8	
B2-9	
B2-10	
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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL TEST (Continued)

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

CORE ALTERATION

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

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

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

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

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

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

OPERATIONAL CONDITION

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

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

b. All equipment hatches are closed and sealed.

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

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

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

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formula, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wasts based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

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 1.04 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 1.04 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

BASES

2.0 The fuel cladding, reactor pressure vessel, and primary system 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 1.04. MCPR > 1.04 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 NRC approved CPR correlation may not be 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 x 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 x 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 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 an approved critical power correlation. Details of the fuel cladding integrity safety limit calculation are given in Reference 1 and 2.

Uncertainties used in the determination of the fuel cladding integrity safety limit and the bases of these uncertainties are presented in Reference 1 and 2.

The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Brunswick Unit 1 during any fuel cycle could not be as severe as the distribution used in the analysis. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

References

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, Revision 8.

2. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, Amendment 14.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. However, the pressure safety limit is set high enough such that no foreseeable circumstances can cause the system pressure to rise to this limit. The pressure safety limit is also selected to be the lowest transient overpressure allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III and USAS Piping Code, Section B 31.1.

2.1.4 REACTOR VESSEL WATER LEVEL

With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety Limit has been established at the top of the active irradiated fuel to provide a point which can be monitored and also provide an adequate margin for effective action.

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 is 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 in Section 7.5 of the FSAR. The most severe case involves an initial condition in which the reactor is just subcritical and the IRMs 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 shut down and peak power is limited to 1% of RATED THERMAL POWER, thus maintaining MCPR above 1.04. 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 an 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 the possible sources of reactivity

2. Average Power Range Monitor (Continued)

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 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 pressure measurement compared to the highest pressure that

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES (Continued)

3. Reactor Vessel Steam Dome Pressure-High (Continued)

occurs in the system during a transient. This setpoint is effective at low power/flow conditions when the turbine stop valve closure is bypassed. For a turbine trip under these conditions, the transient analysis indicates a considerable margin to the thermal hydraulic limit.

4. Reactor Vessel Water Level-Low, Level #1

The reactor water level trip point was chosen far enough below the normal operating level to avoid spurious scrams but high enough above the fuel to assure that there is adequate water to account for evaporation losses and displacement of cooling following the most severe transients. This setting was also used to develop the thermal-hydraulic limits of power versus flow.

5. Main Steam Line Isolation Valve-Closure

The low-pressure isolation of the main steamline trip was provided to give protection against rapid depressurization and resulting cooldown of the reactor vessel. Advantage was taken of the shutdown feature in the run mode, which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low pressures does not occur. Thus, the combination of the low-pressure isolation and isolation valve closure reactor trip with the mode switch in the Run position assures the availability of neutron flux protection over the entire range of the Safety Limits. In addition, the isolation valve closure trip with the mode switch in the Run position anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure.

6. Main Steam Line Radiation - High

The Main Steam Line Radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a scram is initiated to reduce the continued failure of fuel cladding. At the same time, the Main Steam Line Isolation Valves are closed to limit the release of fission products. The trip setting is high enough above background radiation level to prevent spurious scrams, yet low enough to promptly detect gross failures in the fuel cladding.

7. Drywell Pressure, High

High pressure in the drywell could indicate a break in the nuclear process systems. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trips.

8. Scram Discharge Volume Water Level-High

The scram discharge tank receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this tank fill up to a point where there is insufficient volume to accept the displaced water, control rod movement would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods when they are tripped.

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 10% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed.

10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low

The reactor protection initiates a scram signal after the control valve hydraulic oil pressure decreases due to a load rejection exceeding the capacity of the bypass valves or due to hydraulic oil system rupture. The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where upon a loss of oil pressure, control valves closure time is approximately twice as long as that for the stop valves, which means that resulting transients, while similar, are less severe than for stop valve closure. No fuel damage occurs, and reactor system pressure does not exceed the safety relief valve setpoint. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first-stage pressure.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

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 1.04 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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the following limits:

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

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5, initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

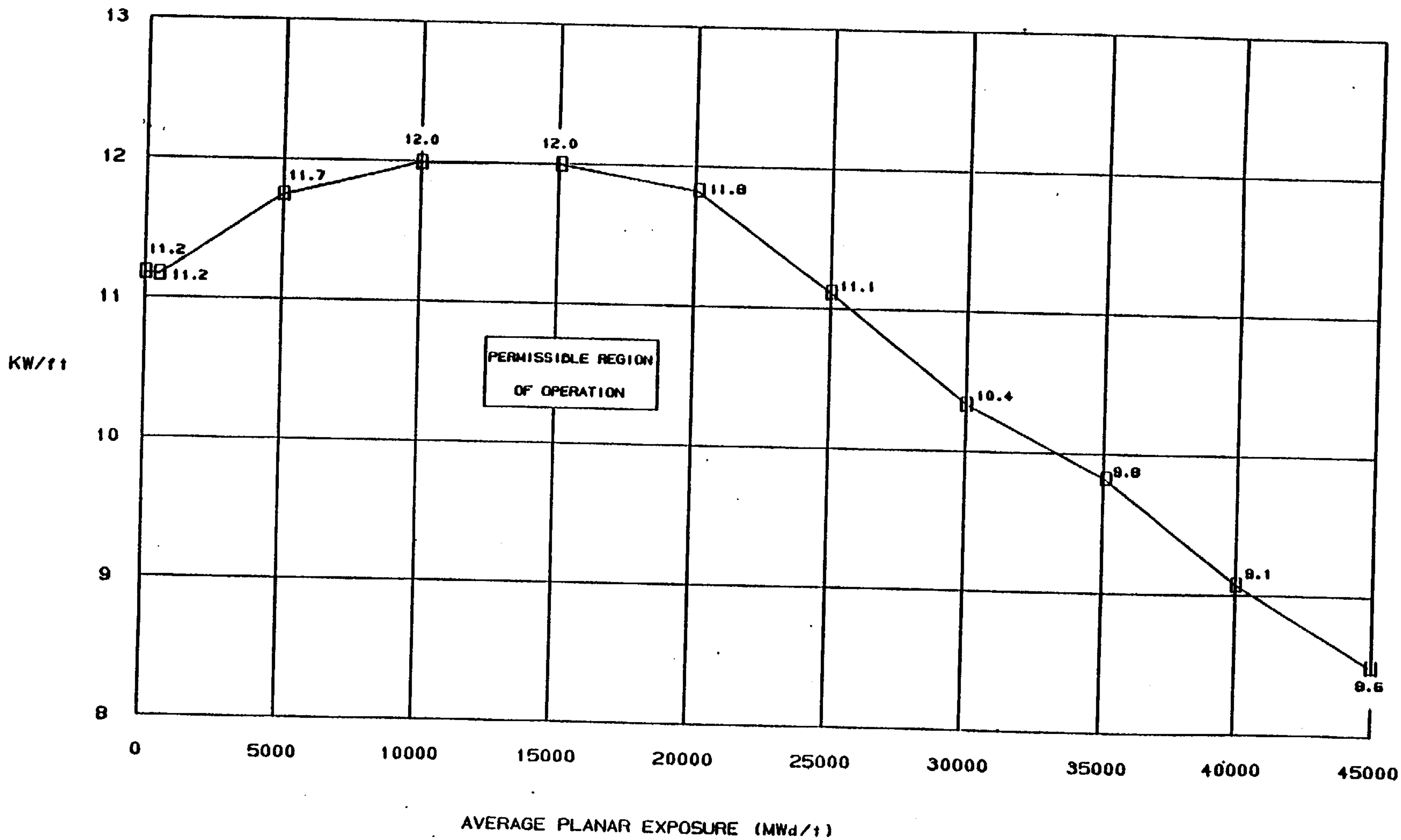
4.2.1 All APLHGRs shall be verified to be equal to or less than the applicable limit determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5:

a. At least once per 24 hours,

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

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

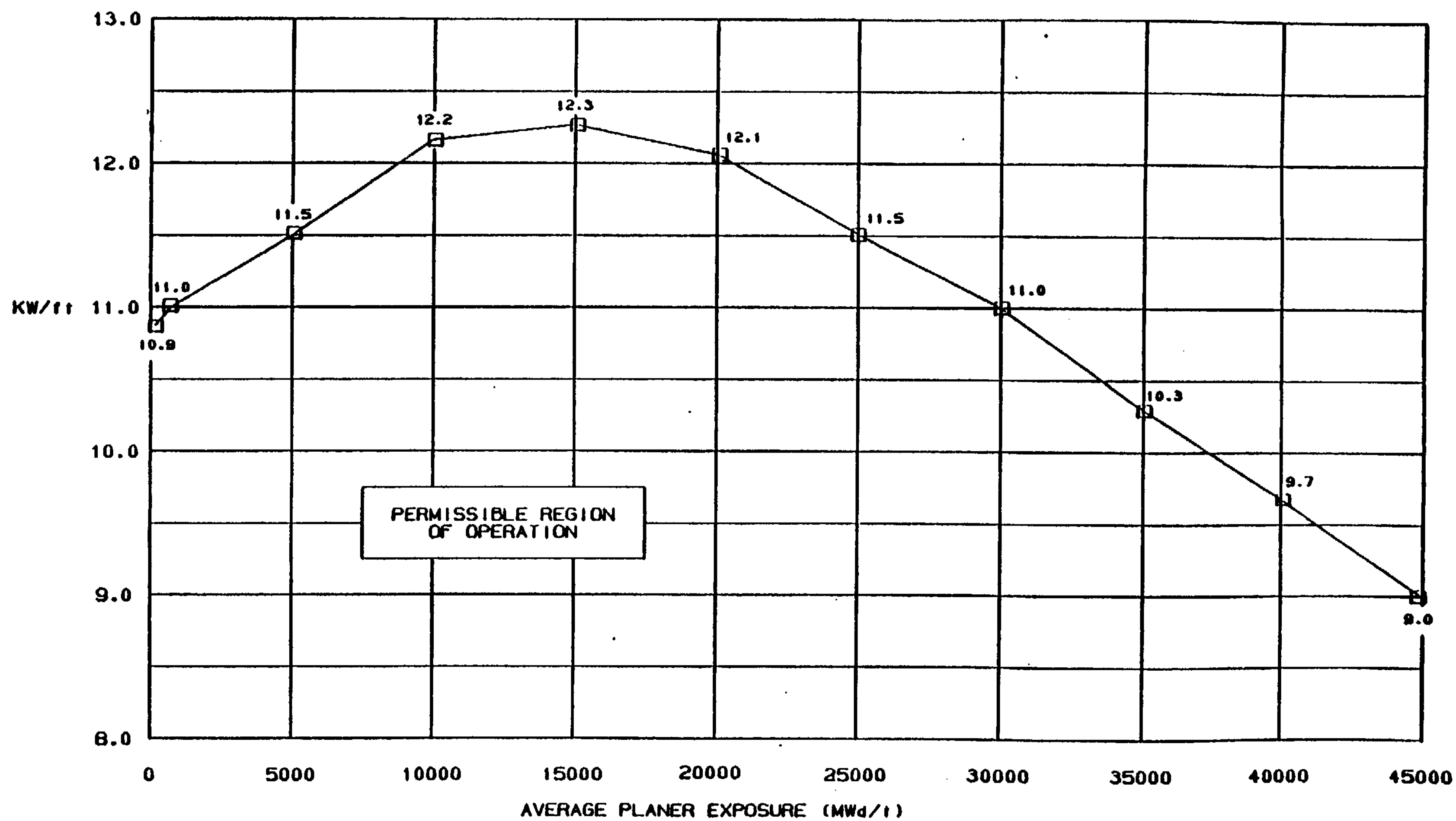
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE



FUEL TYPE P8DRB284H (P8X8R)

Figure 3.2.1-1

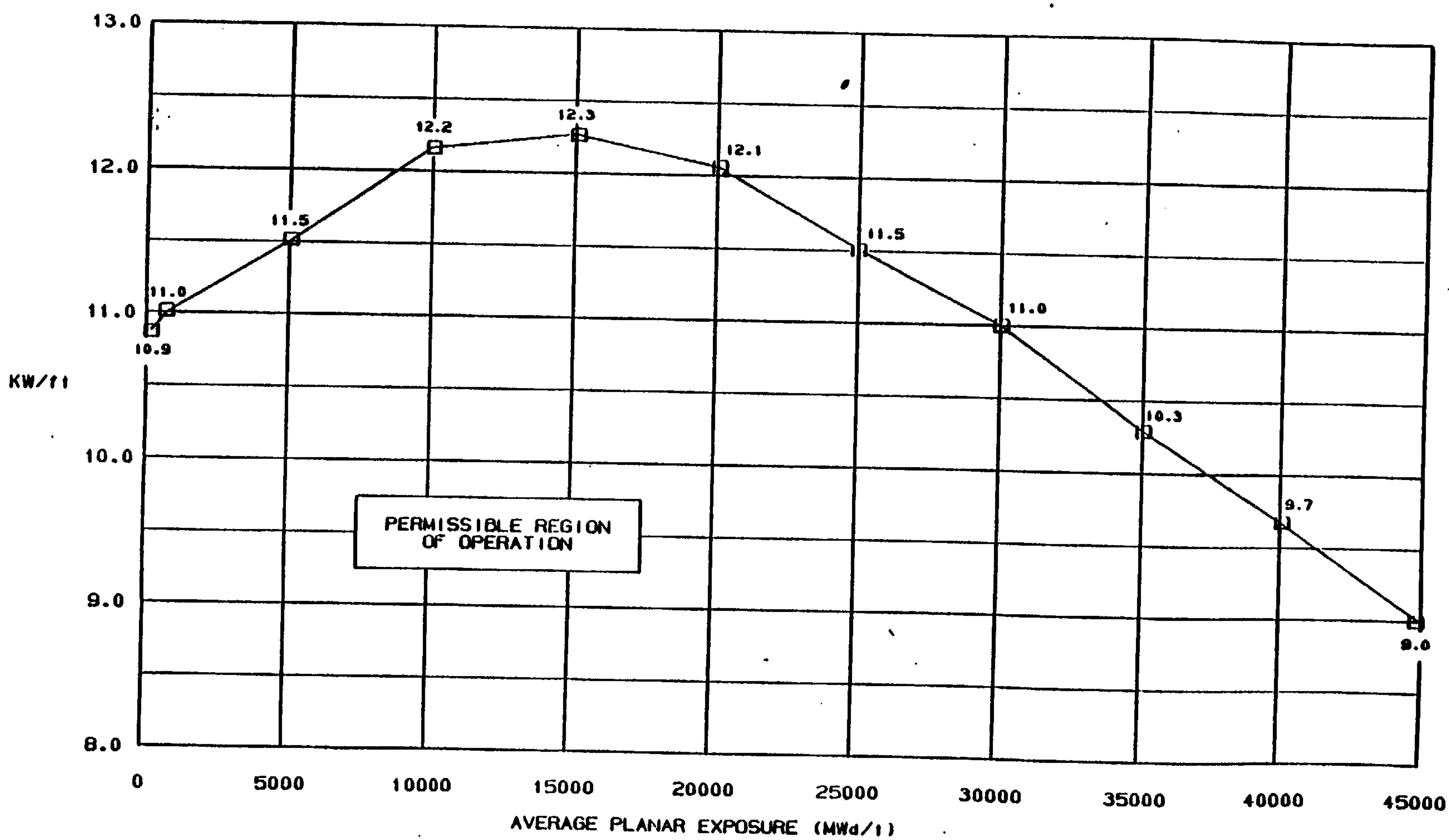
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE



FUEL TYPE P8DRB299 (P8XBR)

Figure 3.2.1-2

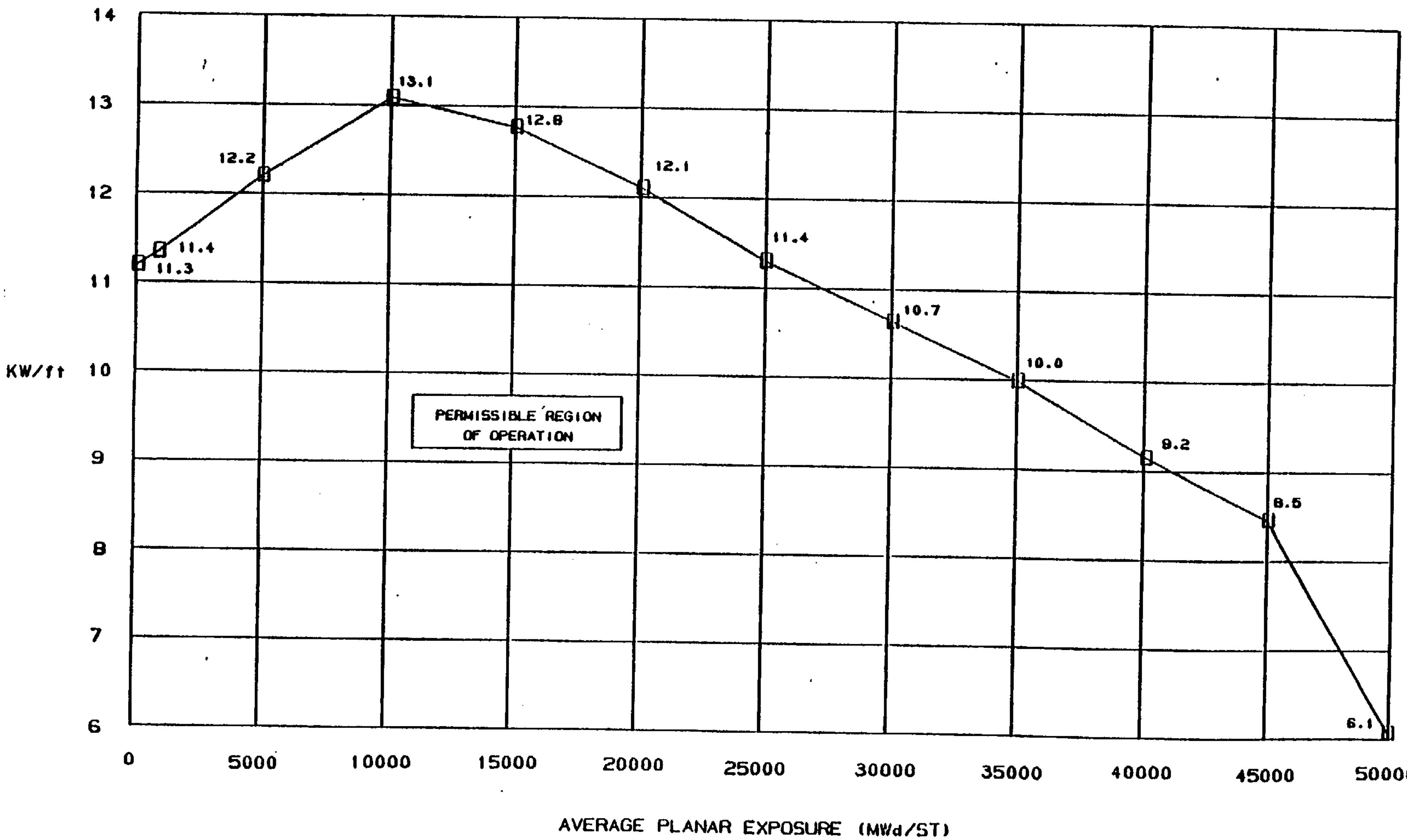
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE



FUEL TYPE BP8DRB299 (BP8XBR)

Figure 3.2.1-3

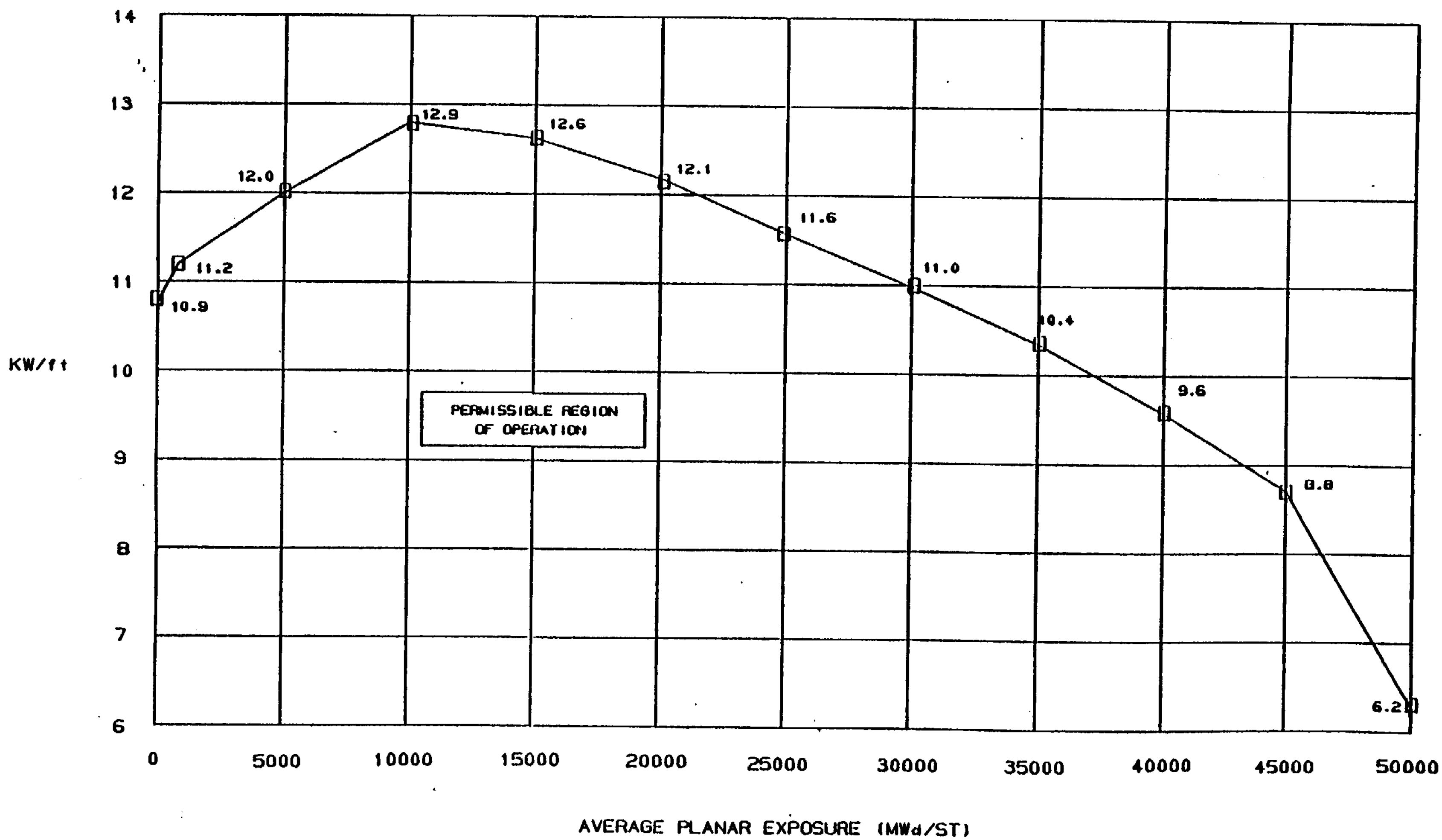
NOTE: THIS CURVE REPRESENTS THE MOST LIMITING APLHGR TO BE USED FOR HAND CALCULATIONS. THE LIMITING VALUES FOR EACH LATTICE ARE IN THE CORE MONITORING SYSTEM.



FUEL TYPE 8D323B (GE8)

Figure 3.2.1-4

NOTE: THIS CURVE REPRESENTS THE MOST LIMITING APLHGR TO BE USED FOR HAND CALCULATIONS. THE LIMITING VALUES FOR EACH LATTICE ARE IN THE CORE MONITORING SYSTEM.



FUEL TYPE BD339A (GE8)

Figure 3.2.1-5

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The flow-biased APRM scram trip setpoint (S) and rod block trip set point (S_{RB}) shall be established according to the following relationship:

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

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

where: S and S_{RB} are in percent of RATED THERMAL POWER.
W = Loop recirculation flow in percent of rated flow,
T = Lowest value of the ratio of design TPF divided by the MTPF obtained for any class of fuel in the core ($T \leq 1.0$), and

Design TPF for: P8 x 8R fuel = 2.39
BP8 x 8R fuel = 2.39
GE8 fuel = 2.48

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

ACTION:

With S or S_{RB} exceeding the allowable value, initiate corrective action within 15 minutes and continue corrective action so that S and S_{RB} are within the required limits within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The MTPF for each class of fuel shall be determined, the value of T calculated, and the flow biased APRM trip setpoint adjusted, as required:

a. At least once per 24 hours,

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

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

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3.1 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, shall be equal to or greater than the MCPR limit times the K_f shown in Figure 3.2.3-1 with the following MCPR limit adjustments:

a. Beginning-of-cycle (BOC) to end-of-cycle (EOC) minus 2000 MWD/t with ODYN OPTION A analyses in effect, the MCPR limits are listed below:

1. MCPR for P8 x 8R fuel = 1.32
2. MCPR for BP8 x 8R fuel = 1.32
3. MCPR for GE8 fuel = 1.32

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

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

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

1. MCPR for P8 x 8R fuel = 1.25
2. MCPR for BP8 x 8R fuel = 1.25
3. MCPR for GE8 fuel = 1.25

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

1. MCPR for P8 x 8R fuel = 1.30
2. MCPR for BP8 x 8R fuel = 1.30
3. MCPR for GE8 fuel = 1.30

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

ACTION:

With MCPR, as a function of core flow, less than the applicable limit determined from Figure 3.2.3-1 initiate corrective action within 15 minutes and restore MCPR to within the applicable limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

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

LIMITING CONDITION FOR OPERATION

3.2.3.2 For the OPTION B MCPR limits listed in specification 3.2.3.1 to be used, the cycle average 20% (notch 36) scram time (τ_{ave}) shall be less than or equal to the Option B scram time limit (τ_B), where τ_{ave} and τ_B are determined as follows:

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

i = Surveillance test number,
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test, and
τ_i = Average scram time to notch 36 for surveillance test i

$$\tau_B = \mu + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i}\right)^{1/2} (\sigma), \text{ where:}$$

i = Surveillance test number
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test
N_1 = Number of rods tested at BOC,
μ = 0.813 seconds
(mean value for statistical scram time distribution from de-energization of scram pilot valve solenoid to pickup on notch 36),
σ = 0.018 seconds
(standard deviation of the above statistical distribution).

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

LIMITING CONDITIONS FOR OPERATION (Continued)

ACTION:

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

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

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

where: τ_A = 1.05 seconds, control rod average scram insertion time limit to notch 36 per Specification 3.1.3.3,
$MCPR_{option\ A}$ = Determined from Table 3.2.3.2-1,
$MCPR_{option\ B}$ = Determined from Table 3.2.3.2-1, or,

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

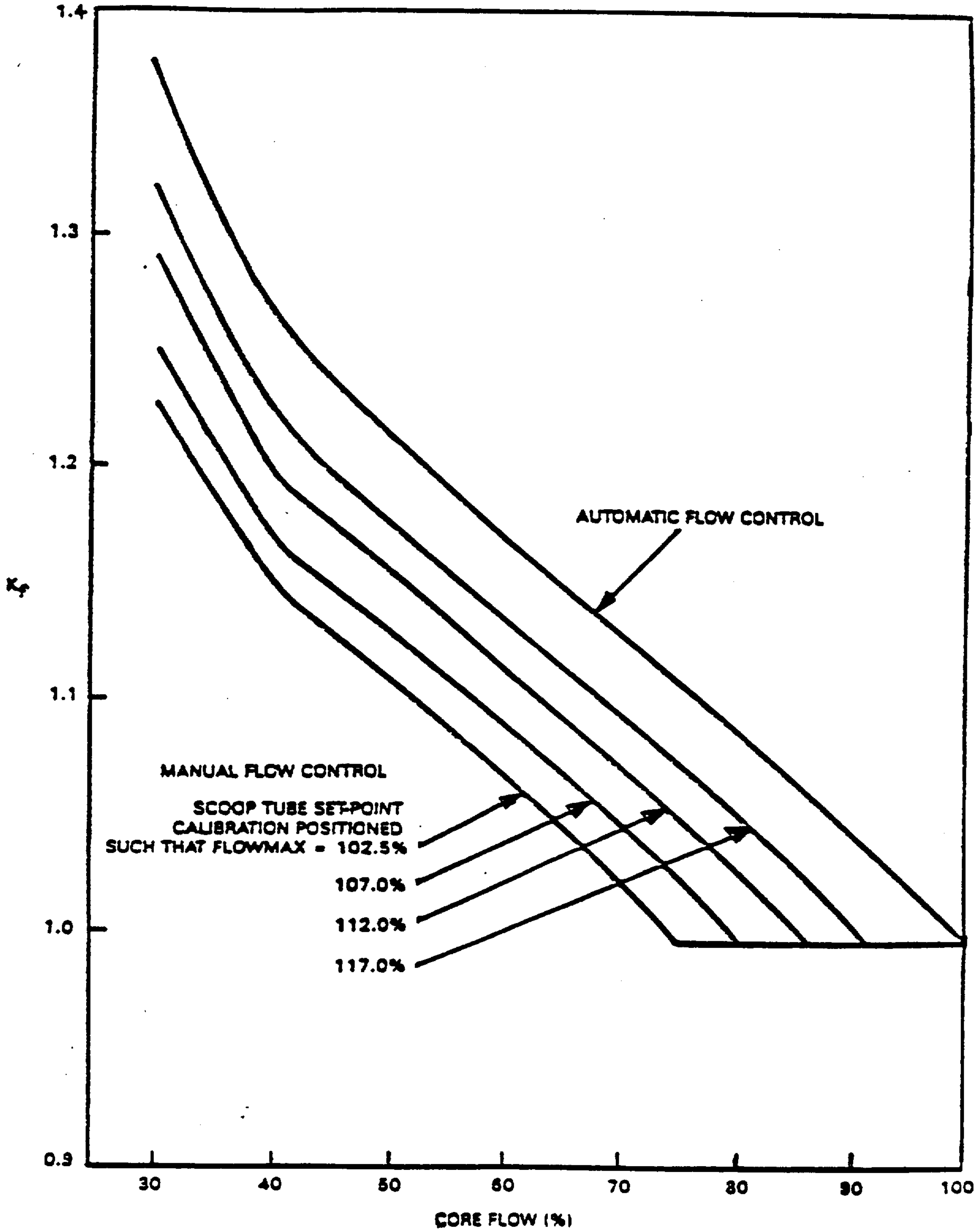
SURVEILLANCE REQUIREMENTS

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

TABLE 3.2.3.2-1

TRANSIENT OPERATING LIMIT MCPR VALUES

TRANSIENT	FUEL TYPE P8x8R		BP8x8R		GE8	
NONPRESSURIZATION TRANSIENTS						
BOC → EOC	1.25		1.25		1.25	
PRESSURIZATION TRANSIENTS						
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
BOC → EOC - 2000	1.32	1.25	1.32	1.25	1.32	1.25
EOC - 2000 → EOC	1.34	1.30	1.34	1.30	1.34	1.30



K_f FACTOR

FIGURE 3.2.3-1

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed 13.4 kw/ft for P8 x 8R and BP8 x 8R fuel assemblies and 14.4 kw/ft for GE8 fuel assemblies.

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

ACTION:

With the LHGR of any fuel rod exceeding the above limit, initiate corrective action within 15 minutes and continue corrective action so that the LHGR is within the limit within 4 hours, or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

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

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

INSTRUMENTATION

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 1.04 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 3.3.4-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. APRM (C51-APRM-CH. A,B,C,D,E,F)		
a. Upscale (Flow Biased)	$\leq$ (0.66W + 42%)T[(a)]	$\leq$ (0.66W + 42%)T[(a)]
b. Inoperative	NA	NA
c. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
d. Upscale (Fixed)	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 12% of RATED THERMAL POWER
2. ROD BLOCK MONITOR (C51-RBM-CH.A,B)		
a. Upscale	$\leq$ (0.66W + 41%)T[(a)]	$\leq$ (0.66W + 41%)T[(a)]
b. Inoperative	NA	NA
c. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
3. SOURCE RANGE MONITORS (C51-SRM-K600A,B,C,D)		
a. Detector not full in	NA	NA
b. Upscale	$\leq 1 \times 10^5$ cps	$\leq 1 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3 cps	$\geq$ 3 cps
4. INTERMEDIATE RANGE MONITORS (C51-IRM-K601A,B,C,D,E,F,G,H)		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 of full scale	$\leq$ 108/125 of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
5. SCRAM DISCHARGE VOLUME (C11-LSH-N013E)		
a. Water Level - High	$\leq$ 73 gallons	$\leq$ 73 gallons

[(a)] T as defined in Specification 3.2.2.

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 MPCR from becoming less than 1.04 during the limiting power transient analyzed in Section 15 of the Updated 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 1.04. 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 temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod-power distribution within an assembly. The peak clad temperature is calculated assuming the LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5.

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, and 3.2.1-5 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, and 3.2.1-5, (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and countercurrent flow limitation as described in Reference 2, are included in the reflooding calculations.

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

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

FOR BRUNSWICK-UNIT 1

Plant Parameters:

Core Thermal Power . . . . 2531 Mwt which corresponds 105% of rated steam flow*

Vessel Steam Output. . . . . .10.96 x 10^6 Lbm/h which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure . . . . . 1055 psia

Recirculation Line
Break Area for Large Breaks
a. Discharge — 2.4 ft^2 (DBA); 1.9 ft^2 (80% DBA)

b. Suction — 4.2 ft^2

Number of Drilled Bundles — 560

Fuel Parameters:

FUEL TYPES	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO**
Reload Core	BP/P8 x 8R	13.4	1.4	1.2
	GE8	14.4	1.4	1.2

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

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

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

POWER DISTRIBUTION LIMITS

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 2.39 for P8 x 8R and BP8 x 8R fuel and 2.48 for GE8 fuel. The scram setting and rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.0 in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and peak flux indicates a TOTAL PEAKING FACTOR greater than 2.39 for P8 x 8R and BP8 x 8R fuel and 2.48 for GE8 fuel. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM high flux scram curve by the reciprocal of the APRM gain change. The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit 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 1.04, 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 required minimum operating limit MCPR of Specification 3.2.3 is obtained when the transient which yields the largest ΔCPR is added to the Safety Limit MCPR of 1.04. Prior to analysis of abnormal operational transients, an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multichannel steady state flow distribution model as described in Section 4.4 of NEDO-20360[4] and on core parameters shown in Reference 3, response to Items 2 and 9.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.124 TO FACILITY OPERATING LICENSE NO. DPR-71

CAROLINA POWER & LIGHT COMPANY, et al.

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

DOCKET NO. 50-325

1.0 INTRODUCTION

By letter from L. W. Eury, Carolina Power & Light Company (CP&L), dated August 1, 1988 (Ref. 1), Technical Specifications (TS) changes were proposed for the operation of the Brunswick Steam Electric Plant, Unit 1, for Cycle 7. In support of these changes, the submittal included the General Electric (GE) Reports "Supplemental Reload Licensing Report for Brunswick Steam Electric Plant Unit 1, Cycle 7" (Ref. 2) and "Loss-of-Coolant Analysis for Brunswick Steam Electric Plant, Unit 1 " (Ref. 3).

The reload for Cycle 7 is generally a normal reload with no unusual core features or characteristics. The Technical Specifications changes primarily relate to the inclusion of new and/or revised Minimum Critical Power Ratio (MCPR) limits, Average Power Range Monitor (APRM) setpoints, Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits, Linear Heat Generation Rate (LHGR) limits for all of the fuel using Cycle 7 core and transient parameters. The new fuel is the extended burnup type that has been used in several recent GE reloads.

Environmental related information on extended fuel irradiation was provided previously by letter dated September 25, 1987.

2.0 EVALUATION

2.1 Reload Description

The Brunswick 1, Cycle 7 reload will retain 16 P8x8R and 332 BP8x8R GE fuel assemblies from the previous cycle and add 184 new GE8x8EB fuel assemblies. The reload is based on a previous cycle core nominal average exposure of 21,072 megawatt days per metric ton (MWD/MT) and Cycle 7 end of cycle (EOC) exposure of 21,230 MWD/MT. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery. This loading is acceptable.

2.2 Fuel Design

The new fuel for Cycle 7 is the GE extended burnup fuel GE8x8EB. The fuel designations are BD339A and BD323B. This fuel type has been approved in the NRC Safety Evaluation Report for Amendment 10 to GESTAR II (Refs. 4

and 5). The specific descriptions of this fuel have been submitted in Amendment 18 to GESTAR II. The specific descriptions of this fuel are presented for Brunswick 1 in Reference 3. These fuel descriptions are acceptable.

The proposed Linear Heat Generation Rate (LHGR) for the GE8x8EB fuel is 14.4 kw/ft as compared to 13.4 kw/ft for the other GE fuel. This LHGR has been reviewed and accepted for this fuel in the GE extended burnup fuel review (Ref. 4). This LHGR limit is acceptable for the new fuel in Cycle 7.

2.3 Nuclear Design

The nuclear design analyses for Cycle 7 have been performed by GE with the approved methodology described in GESTAR II (Ref. 5). The results of these analyses are given in the GE reload report (Ref. 2) in standard GESTAR II format. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 1.2% and 1.0% delta K at both beginning of cycle (BOC) and at the exposure of minimum shutdown margin, respectively; thus fully meeting the required 0.38% delta K.

The standby liquid control system also meets shutdown requirements with a shutdown margin of 3.6% delta K. Since these and other Cycle 7 nuclear design parameters have been obtained using previously approved methods and fall within expected ranges, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design analyses for Cycle 7 have been performed by GE with the approved methodology described in GESTAR II (Ref. 5) and the results are given in the GE reload report (Ref. 2). The parameters used for the analyses are those approved in Reference 5 for the Brunswick class BWR 4 except for the parameters listed in Appendix C of Reference 2. The GEMINI system of methods (approved in Ref. 6) was used for relevant transient analyses.

The approved GEXL-PLUS CPR correlation was used in developing the operating and safety limit MCPRs (Refs. 7 and 8). For off-nominal flow conditions the operating limit MCPR values are adjusted with a Kf MCPR multiplier. This multiplier has been revised to reflect the use of the GEXL-PLUS CPR correlation.

The Operating Limit MCPR (OLMCPR) values are determined by the limiting transients, which are usually Rod Withdrawal Error (RWE), Feedwater Controller Failure (FWCF), Loss of Feedwater Heating (LFH), and Load Rejection Without Bypass (LRWBP). The analyses of these events, using the ODYN option A and B approaches for pressurization transients, provide new Cycle 7 Technical Specification values of OLMCPR in the standard operating region.

For Cycle 7, Brunswick 1 has elected, following standard practice, to have exposure dependent OLMCPR values. Two exposure regions were analyzed: (1) Beginning of Cycle (BOC) to End of Cycle minus 2 Gigawatt days per Short Ton (EOC- 2GWD/ST) and (2) EOC-2GWD/ST to EOC. For standard operating conditions, the LRWBP event is controlling at both Option A and B limits. These OLMCPR results are reflected in Technical Specification changes. Approved methods (Ref. 5) were used to analyze these events (and others which could be limiting) and the analyses and results are acceptable and fall within expected ranges.

The Safety Limit MCPR (SLMCPR) is set so that less than 0.1 % of the fuel pins in the core are subject to boiling transition when some fuel in the core is at the SLMPCR. The SLMPCR is being changed to 1.04. This change has been approved by the NRC for D-lattice cores operating with the second successive reload cores of P8x8R, BP8x8R, GE8x8E or GE8x8EB fuel types with high bundle R-factor (Ref. 9). Brunswick Unit 1 is such a D-lattice plant with Cycle 7 being the third successive reload core with high bundle R-factor fuel. In addition, the staff found a similar change for Brunswick Unit 2 acceptable (Ref. 10). Thus, the change to the SLMCPR is acceptable.

The mean and standard deviations of the control rod scram speed data that are used to compute the adjusted mean scram time (tau) are being changed. This change revises the values for the constants mu and sigma used to calculate the ODYN Option B scram time limit, which is used to select the applicable OLMCPR. The revised values conform to the approved GEMINI/ODYN analysis methods. They are appropriate for the insertion time requirements where control rod notch position 36 corresponds to the 20% scram time position. These changes lead to a conservative tau and OLMCPR and are, therefore, acceptable.

The Brunswick Units 1 and 2 Technical Specifications have requirements for the detection and suppression of core thermal-hydraulic instability for two or one recirculation loop operation (Ref. 11). These specifications reflect the conclusions of the staff Generic Letters 86-02 and 86-09 (Refs. 12 and 13), which were based on extensive stability reviews and the recommendations of the GE report SIL-380 (Ref. 14). Recently, LaSalle Unit 2 experienced excessive neutron flux oscillations while in natural circulation after a dual recirculation pump trip. After investigation of this event the NRC staff has identified generic safety implications regarding power oscillations in Boiling Water Reactors. NRC Bulletin No. 88-07 (Ref. 15), dealing with this subject, was issued. The licensee has responded to the Bulletin action items by Reference 16. Brunswick Units 1 and 2 have procedures and operator training programs in place to address uncontrolled power oscillations. In addition, plant procedures were revised to address the LaSalle event. The licensee further stated that the training program was revised to make operators more aware of the consequences of operating in the region of thermal hydraulic instability, and to emphasize the need to manually scram the reactor if power oscillations are not promptly terminated.

The action items of Bulletin No. 88-07 are being addressed generically and will be reviewed under an NRC Regional Office inspection in accordance with a Temporary Instruction procedure.

2.5 Transient and Accident Analyses

The transient and accident analysis methodologies used for Cycle 7 are described in GESTAR II (Ref. 5). The GEMINI system of methods (Ref. 6) option was used for transient analyses. The limiting MCPR events for Brunswick 1, Cycle 7, are indicated in Section 2.4. The core wide transient analysis methodologies and results are acceptable and fall within expected ranges.

The RWE was analyzed on a plant and cycle specific basis (as opposed to the statistical approach) and a rod block setpoint of 107 was selected to provide an OLMCPR of 1.18 for all fuel types. The fuel misorientation event was analyzed with standard methods for the D lattice fuel, giving a nonlimiting MCPR of 1.25.

The results of the cycle specific control rod drop accident from both cold conditions and hot standby conditions meet the NRC acceptance criterion (280 calories per gram peak enthalpy) for this event. The local transient event analyses have been performed with approved methods and acceptable input assumptions and result in acceptable consequences for Cycle 7.

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

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

2.6 Technical Specifications

The Technical Specifications changes for Cycle 7 are as follows:

(a) The Minimum Critical Power Ratio (MCPR) safety limit has been revised to 1.04. This is incorporated in TS 3.1.4.3.a.3 and 3.3.4.b.3. We find this is acceptable.

(b) MAPLHGR limits are provided for the new fuel. The changes are to TS 3/4.2.1, Figure 3.2.1-4 and Figure 3.2.1-5 and are acceptable.

(c) The revision of the K_f curve, Figure 3.2.3-1, is acceptable.

(d) A 14.4 kw/ft LHGR limit for GE8 fuel has been implemented. The changes are to TS 3.2.4 and are acceptable.

(e) New APRM scram setpoints and rod block setpoints based on new design total peaking factors for GE8 fuel have been provided. The changes are to TS 3.2.2 and Table 3.3.4-2 and are acceptable.

(f) Changes to TS 3.2.3.1 and Table 3.2.2.2-1, which incorporate new MCPR limits for GE8 fuel, are acceptable.

(g) The revised values of mu and sigma and the reference to notch 36 have been incorporated in TS 3.2.3.2 and are acceptable.

There are also minor administrative changes to the index, pagination, the definitions of Critical Power Ratio and Physics Tests, associated Bases, and references. These are all acceptable.

2.7 EXTENDED BURNUP

The licensee has requested authorization to allow Unit 1 burnup to 60,000 MWD/MT. The staff and licensee evaluated the potential impact of this change in a previous licensing action associated with Unit 2. The evaluation can be found in a Safety Evaluation issued to the licensee on September 20, 1988. The results of the evaluation for Unit 2 are equally applicable for Unit 1 because the accident analyses for Brunswick apply to both units.

The staff concludes that the only potential increased dose potentially resulting from a design basis accident with extended fuel burnup to 60,000 MWD/MT is the thyroid dose resulting from fuel handling accidents. The small increase is insignificant, in that the doses remain well within the 300 Rem thyroid exposure guideline values of 10 CFR Part 100.

3.0 SUMMARY

The staff has reviewed the reports submitted for the Cycle 7 operation of Brunswick Unit 1. Based on this review, the staff concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle. Lastly, extended fuel burnup to 60,000 MWD/MT is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on January 31, 1989 at 54 FR 4924. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration, which was published in the Federal Register on November 30, 1988 at 53 FR 48325, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina had no comments.

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

References: Attached

Principal Contributor: D. Katze
E. Tourigny

Dated: February 6, 1989

REFERENCES

1. Letter from L. W. Eury (CP&L) to NRC, "Brunswick Steam Electric Plant Unit No.1 Request for License Amendment, Fuel Cycle No. 7- Reload Licensing," dated August 1, 1988.

2. GE Report 23A5896, "Supplemental Reload Licensing Report for Brunswick Steam Electric Plant, Unit 1, Reload 6, Cycle 7," June 1988.

3. GE Report NEDE-24165-P, "Loss-Of-Coolant Accident Analysis Report for Brunswick Steam Electric Plant, Unit 1, " April 1988.

4. Letter (and attachment) from C. Thomas (NRC) to J. Charnley (GE)," Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A-6, Amendment 10," dated May 28, 1985.

5. GESTAR II, NEDE-24011, Revision 8, "General Electric Standard Application for Reactor Fuel," May 1986.

6. Letter (and attachment) from G. Lainas (NRC) to J. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, GE Generic Licensing Reload Report, Supplement to Amendment 11," dated March 22, 1986.

7. Letter from A. C. Thadani (NRC) to J. Charnley (GE)," Acceptance for Referencing of Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application For Reactor Fuel," dated March 14, 1988.

8. Letter from A. C. Thadani (NRC) to J. Charnley (GE)," Acceptance for Referencing of Application of Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application For Reactor Fuel," dated May 5, 1988.

9. Letter from Ashok C. Thadani (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Amendment 14 to General Electric Licensing Topical Report NEDE-24011-P-A,'General Electric Standard Application for Reactor Fuel,' " dated December 27, 1987.

10. Letter (and attachment) from E. Sylvester, (NRC) to E. E. Utley (CP&l), "Issuance or Amendment No. 151 to Facility Operating License No. DPR-62- Brunswick Steam Electric Plant, Unit 2, Regarding Upgraded MCPR Safety Limit, Cycle 8" dated April 12, 1988.

11. Letter from M. W. Hodges to E. G. Adensam, "SER for Core Thermal Hydraulic Stability Technical Specification Changes for Brunswick 1 and 2," dated July 16, 1987.

12. Generic Letter No. 86-02, "Technical Resolution of Generic Issue B-19 Thermal Hydraulic Stability," January 23, 1986.

13. Generic Letter No. 86-09, "Technical Resolution of Generic Issue No. B-59-(N-1) Loop Operation in BWRs and PWRS," March 31, 1986.

14. General Electric Service Information Letter No. 380, Revision 1, February 10, 1984.

15. NRC Bulletin No. 88-07, "Power Oscillations In Boiling Water Reactors," June 15,1988.

16. Letter from M. A. McDuffie (CP&L) to NRC, "Brunswick Steam Electric Plant Unit Nos. 1 and 2. "Response to NRC Bulletin No. 88-07, "Power Oscillations In Boiling Water Reactors," dated September 15, 1988.