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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 151
License No. DPR-62

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

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated February 3, 1988, as supplemented March 30, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 151, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 12, 1988

LA:PD21:DRPR	PM:PD21:DRPR	PE:PD21:DRPR	OGC-B	D:PD21:DRPR
PAnderson/clh	ESylvester	BMozafari	MYoung	EAdensam
4/8/88	4/8/88	4/8/88	4/6/88	4/11/88

ATTACHMENT TO LICENSE AMENDMENT NO. 151

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

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

Remove Pages	Insert Pages
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	B2-8
B2-9	-
B2-10	-
B2-11	-
B2-12	-
B2-13	-
3/4 1-17	3/4 1-17
3/4 2-8	3/4 2-8
3/4 2-12	3/4 2-12
B3/4 1-2	B3/4 1-2
B3/4 2-3	B3/4 2-3

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 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 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.

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

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 2 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.

Reference

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (latest approved revision).

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.

2.2.1 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

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 THEMAL 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 an adequate margin for the Safety Limits and yet allows an operating margin that reduces the possiblility 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

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

pressure measurement compared to the highest pressure that 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 steam line 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 levels 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

Low turbine control valve hydraulic pressure will initiate the Select Rod Insert function and the preselected group of control rods will be fully inserted. Select Rod Insert is an operational aid designed to insert a predetermined group of control rods immediately following either a generator load rejection, loss of turbine control valve hydraulic pressure, or by manual operator action using a switch on the R-T-G board. The assignment of control rods to the Select Rod Insert function is based on the start-up and fuel warranty service associated with each control rod pattern, on RCS considerations, and on a dynamic function of both time and core patterns.

Approximately ten percent of the control rods in the reactor will be assigned to the Select Rod Insert function by the operator. This selection will be accomplished by moving the rod scram test switch for those rods from the Normal position to the Select Rod Insert position.

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

Any rod selected for Select Rod Insert shall also have other rods in its notch group selected to ensure that the RSCS criteria of plus-minus one notch position equality is met when the rod pattern is greater than 50% ROD DENSITY and THERMAL POWER $\leq$ 20% of RATED THERMAL POWER. It is possible that a rod pattern within these limits may occur after the Select Rod Insert function operates.

In order to reduce the number of reactor scrams, a 200 millisecond time delay, referenced from the low turbine control valve hydraulic pressure and Select Rod Insert signals, was incorporated to determine turbine bypass valve status via limit switches prior to initiating a reactor scram. If the turbine bypass valves opened in < 200 milliseconds, the reactor scram was bypassed. It was found that during certain reload cycles the MCPR penalties involved with this time delay were more penalizing than the number of scrams saved; therefore, CP&L requested and received NRC approval to set this time at "0" in Amendment No. 14. With the timer set at "0", Select Rod Insert and RPS trip will be initiated simultaneously.

The control valve 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.

REACTIVITY CONTROL SYSTEMS

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 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.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

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

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

   1. MCPR for P8 x 8R fuel = 1.29
   2. MCPR for BP8 x 8R fuel = 1.29
   3. MCPR for GE8 fuel = 1.29

b. EOC minus 2000 MWD/t to EOC with ODYN OPTION A analyses in effect and the end-of-cycle recirculation pump trip system inoperable, the MCPR limits are listed below:

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

c. BOC to EOC minus 2000 MWD/t with ODYN OPTION B analyses in effect and the end-of-cycle recirculation pump trip system inoperable, the MCPR limits are listed below:

   1. MCPR for P8 x 8R fuel = 1.22
   2. MCPR for BP8 x 8R fuel = 1.22
   3. MCPR for GE8 fuel = 1.22

d. EOC minus 2000 MWD/t to EOC with ODYN OPTION B analyses in effect and the end-of-cycle recirculation pump trip system inoperable, the MCPR limits are listed below:

   1. MCPR for P8 x 8R fuel = 1.26
   2. MCPR for BP8 x 8R fuel = 1.26
   3. MCPR for GE8 fuel = 1.26

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

TABLE 3.2.3.2-1

TRANSIENT OPERATING LIMIT MCPR VALUES

TRANSIENT	FUEL TYPE P8x8R		BP8x8R		GE8	
NONPRESSURIZATION TRANSIENTS						
BOC → EOC	1.22		1.22		1.22	
PRESSURIZATION TRANSIENTS						
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
BOC → EOC - 2000	1.29	1.22	1.29	1.22	1.29	1.22
EOC - 2000 → EOC	1.30	1.26	1.30	1.26	1.30	1.26

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 shut down 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 1.04 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 1.04. 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

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of 2.39 for P8x8R and BP8x8R fuel and 2.48 for GE8 fuel. The scram setting and rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.0 in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and peak flux indicates a TOTAL PEAKING FACTOR greater than 2.39 for P8x8R and BP8x8R fuel and 2.48 for GE8 fuel. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM high flux scram curve by the reciprocal of the APRM gain change. The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.04, 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 an instrument trip setting as given in Specification 2.2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

Unless otherwise stated in cycle specific reload analyses, the limiting transient which determines the required steady state MCPR limit is the turbine trip with failure of the turbine bypass. This transient yields the largest Δ MCPR. Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multichannel steady state flow distribution model as described in Section 4.4 of NEDO-20360[4] and on core parameters shown in Reference 3, response to Items 2 and 9.