October 12, 1989

Docket No. 50-324

DISTRIBUTION
See attached sheet

Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2, REGARDING MAXIMUM EXPANDED OF OPERATING DOMAIN (TAC NO. 72853)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 168 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit 2. The amendment consists of changes to the Technical Specifications in response to your March 29, 1989 submittal.

The amendment changes the Technical Specifications to implement an expanded operating domain. The expansion of the operating domain allows the fuel cycle economics to be enhanced through implementation of flow control spectral shift operation and increased core flow coast down capability.

A copy of the Safety Evaluation that relates to both Units 1 and 2 is enclosed. Notice of issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Orignal signed by:

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

Enclosures:
1. Amendment No. 168 to License No. DPR-62
2. Safety Evaluation

cc w/enclosures:
See next page

DF01
1/1

Document Name: BSEP2 AMEND 72853

OFC	LA:PD21:DRPR	PM:PD21:DRPR	D:PD21:DRPR			
NAME	PAnderson	ETourigny:dt	EAdensam			
DATE	9/14/89	9/18/89	10/12/89			

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Mr. L. W. Eury
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

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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. 168
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 March 29, 1989; 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. 168, 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

Orignal signed by:

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: October 12, 1989

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR			
NAME	PAnderson	ETourigny:dt		EAdensam			
DATE	9/14/89	9/18/89	9/26/89	10/12/89			

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ATTACHMENT TO LICENSE AMENDMENT NO. 168

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
I	I
IV	IV
X	X
1-2	1-2
1-3	1-3
2-4	2-4
2-5	2-5
2-6	2-6
B 2-7	B 2-7
B 2-8	B 2-8
3/4 1-17	3/4 1-17
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 3-47	3/4 3-47
3/4 3-49	3/4 3-49
3/4 3-50	3/4 3-50
B 3/4 1-4	B 3/4 1-4
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
6-25	6-25
6-26	6-26

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DEFINITIONS

SECTION

1.0 DEFINITIONS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

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.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.3.1, 6.9.3.2, 6.9.3.3, and 6.9.3.4. Plant operation within these core operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in an assembly which is calculated, by application of an NRC approved CPR 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 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.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to recirculation pump breaker trip from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not be PRESSURE BOUNDARY LEAKAGE.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High[a]	≤ 120 divisions of full scale	≤ 120 divisions of full scale
2. Average Power Range Monitor		
a. Neutron Flux - High, 15%[b]	≤ 15% of RATED THERMAL POWER	≤ 15% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - High[c][d]	≤ (0.66 W + 64%) with a maximum ≤ 113.5% of RATED THERMAL POWER	≤ (0.66 W + 67%) with a maximum ≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - High[d]	≤ 120% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1045 psig	≤ 1045 psig
4. Reactor Vessel Water Level - Low, Level 1	≥ +162.5 inches[g]	≥ +162.5 inches[g]
5. Main Steam Line Isolation Valve - Closure[e]	≤ 10% closed	≤ 10% closed
6. Main Steam Line Radiation - High	≤ 3 x full power background	≤ 3.5 x full power background
7. Drywell Pressure - High	≤ 2 psig	≤ 2 psig
8. Scram Discharge Volume Water Level - High	≤ 109 gallons	≤ 109 gallons
9. Turbine Stop Valve-Closure[f]	≤ 10% closed	≤ 10% closed
10. Turbine Control Valve Fast Closure, Control Oil Pressure-Low[f]	≥ 500 psig	> 500 psig

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

NOTES

(a) The Intermediate Range Monitor scram functions are automatically bypassed when the reactor mode switch is placed in the Run position and the Average Power Range Monitors are on scale.

(b) This Average Power Range Monitor scram function is a fixed point and is increased when the reactor mode switch is placed in the Run position.

(c) The Average Power Range Monitor scram function is varied, Figure 2.2.1-1, as a function of the fraction of rated recirculation loop flow (W) in percent.

(d) The APRM flow-biased simulated thermal power signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

(e) The Main Steam Line Isolation Valve-Closure scram function is automatically bypassed when the reactor mode switch is in other than the Run position.

(f) These scram functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER as measured by turbine first stage pressure.

(g) Vessel water levels refer to REFERENCE LEVEL ZERO.

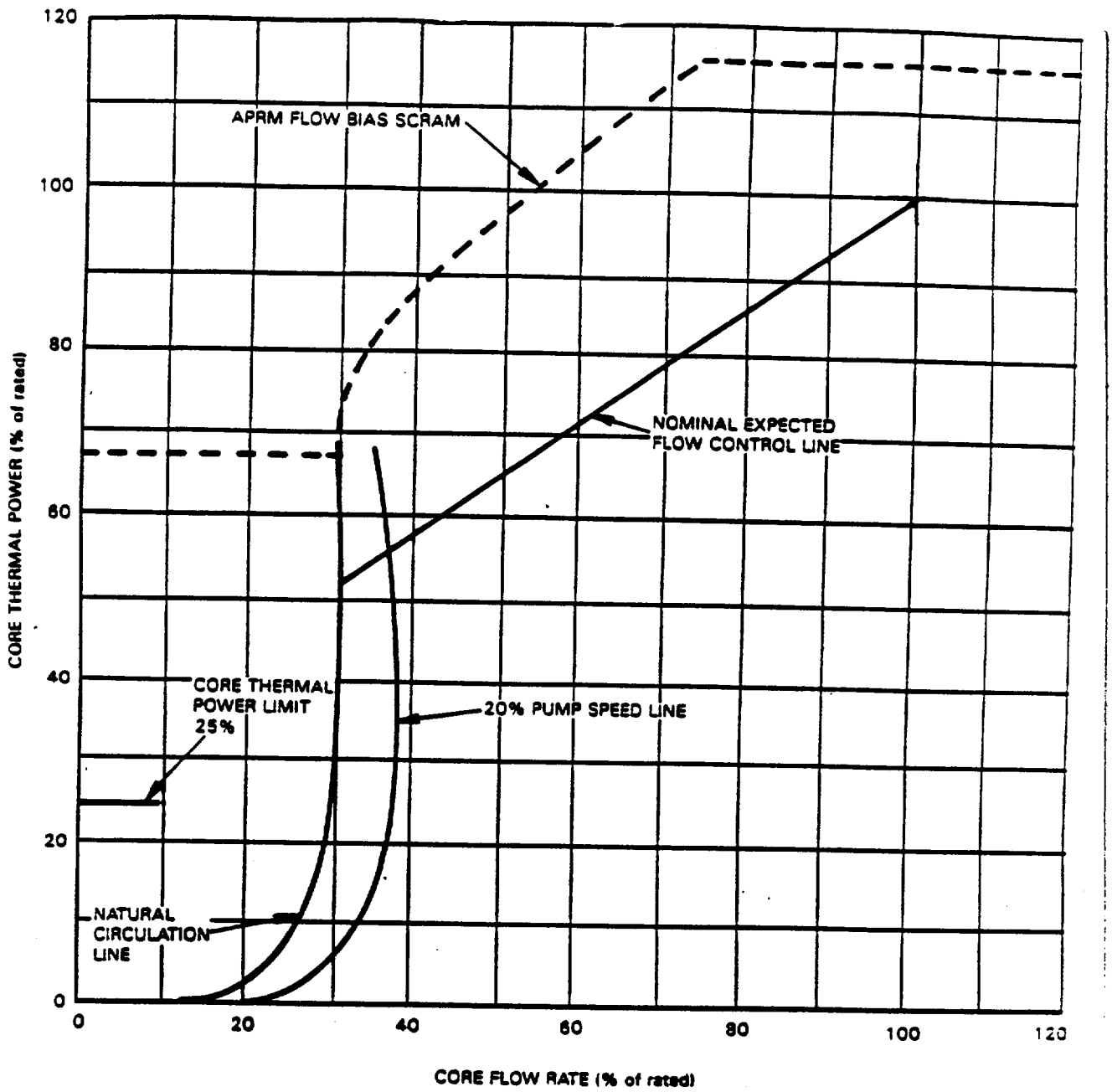

Figure 2.2.1-1. APRM Flow Bias Scram Relationship to Normal Operating Conditions

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 the Safety Limit MCPR of Specification 2.1.2. 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 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

2. Average Power Range Monitor (Continued)

minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the RUN position.

The APRM flux scram trip in RUN mode consists of a flow biased simulated thermal power (STP) scram setpoint and a fixed neutron flux scram setpoint. The APRM flow biased neutron flux signal is passed through a filtering network with a time constant which is representative of the fuel dynamics. This provides a flow referenced signal, e.g., STP, that approximates the average heat flux or thermal power that is developed in the core during transient or steady-state conditions.

The APRM flow biased simulated thermal power scram trip setting at full recirculation flow is adjustable up to the nominal trip setpoint of 113.5% of RATED THERMAL POWER. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 100°F feedwater heating event, than would result with the 120% fixed neutron flux scram trip. The lower flow biased scram setpoint therefore decreases the severity, ΔCPR, of a slow thermal transient and allows lower operating limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the fuel cycle.

The APRM fixed neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow biased simulated thermal power scram.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

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 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 casuing 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. This scram is bypassed when the turbine steam flow is below that corresponding to 30% of RATED THERMAL POWER, as measured by the turbine first-stage pressure.

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.

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.

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

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 that corresponding to 30 percent of RATED THERMAL POWER, 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 with:

a. THERMAL POWER greater than 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER and the MINIMUM CRITICAL POWER RATIO (MCPR) less than 1.70, or

b. THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER and the MCPR less than 1.40.

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.

   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 During power operation, the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall not exceed limits based on applicable APLHGR limit values that have been approved for the respective fuel and lattice type and determined by the approved methodology described in GESTAR-II. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value, adjusted for core flow and core power, for the most limiting lattice (excluding natural uranium) of each type of fuel shown in the applicable figures in the CORE OPERATING LIMITS REPORT.

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 specified in Technical Specification 3.2.1 initiate corrective action within 15 minutes and continue corrective action so that APLHGR is 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

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in Specification 3.2.1:

a. At least once per 24 hours,

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

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

3/4.2.2 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.2.1 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, core power, and cycle average exposure, shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT. The MCPR limits for ODYN OPTION A and ODYN OPTION B analyses, used in the above determination, shall be specified in the CORE OPERATING LIMITS REPORT.

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, core power, and cycle average exposure, less than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT, 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

4.2.2.1 MCPR, as a function of core flow, core power, and cycle average exposure, shall be determined to be equal to or greater than the applicable limit determined of Specification 3.2.2.1:

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.

3/4.2.2 MINIMUM CRITICAL POWER RATIO (ODYN OPTION B)

LIMITING CONDITION FOR OPERATION

3.2.2.2 For the OPTION B MCPR limits provided in the CORE OPERATING LIMITS REPORT 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 ith surveillance test, and
τ_i = Average scram time to notch 36 for surveillance test i

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

i = Surveillance test number
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the ith surveillance test
N_1 = Number of rods tested at BOC,
μ = 0.813 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 specified in the CORE OPERATING LIMITS REPORT 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,

$MCPR_{option\ A}$ = Specified in the CORE OPERATING LIMITS REPORT,
$MCPR_{option\ B}$ = Specified in the CORE OPERATING LIMITS REPORT, or,

b. The OPTION A MCPR limits specified in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.2.2 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.

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.4-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

NOTES

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

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

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

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

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

(f) When (1) THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER and MCPR is less than 1.70, or (2) THERMAL POWER is greater than or equal to 90% of RATED THERMAL POWER and MCPR is less than 1.40.

(g) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(h) This signal is contained in the Channel A logic only.

TABLE 3.3.4-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. APRM		
a. Upscale (Flow Biased)	≤ (0.66W + 58%)[a] with a maximum of ≤ 108% of RATED THERMAL POWER	≤ (0.66W + 61%)[a] with a maximum of ≤ 110% of RATED THERMAL POWER
b. Inoperative	NA	NA
c. Downscale	≥ 3/125 of full scale	≥ 3/125 of full scale
d. Upscale (Fixed)	≤ 12% of RATED THERMAL POWER	≤ 12% of RATED THERMAL POWER
2. ROD BLOCK MONITOR		
a. Upscale	As specified in the CORE OPERATING LIMITS REPORT	As specified in the CORE OPERATING LIMITS REPORT
b. Inoperative	NA	NA
c. Downscale	≥ 94/125 of full scale	NA
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1×10^5 cps	≤ 1×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 3 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 of full scale	≤ 108/125 of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 3/125 of full scale	≥ 3/125 of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level High	≤ 73 gallons	≤ 73 gallons

(a) Where W is the fraction of rated recirculation loop flow in percent.

CONTROL ROD PROGRAM CONTROLS (Continued)

Use of the Banked Position Withdrawal Sequence (BPWS) ensures that in the event of a control rod drop accident, the peak fuel enthalpy will not be greater than 280 cal/gm (Reference 4).

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.6 of the FSAR, Updated and the techniques of the analysis are presented in a topical report (Reference 1) and two supplements (References 2 and 3).

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. The RBM is only required to be operable when the limiting condition described in Specification 3.1.4.3 exists. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods. Further discussion of the RBM system is provided in Reference 5.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for maintaining the reactor subcritical in the event that insufficient rods are inserted in the core when a scram is called for. The volume and weight percent of poison material in solution is based on being able to bring the reactor to the subcritical condition as the plant cools to ambient condition. The temperature requirement is necessary to keep the sodium pentaborate in solution. Checking the volume and temperature once each 24 hours assures that the solution is available for use.

With redundant pumps and a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

1. C. J. Paone, R. C. Stirn, and J. A. Woodley, "Rod Drop Accident Analysis for Large BWRs " G. E. Topical Report NEDO-10527, March 1972.

2. C. J. Paone, R. C. Stirn, and R. M. Yound, Supplement 1 to NEDO-10527, July 1972.

3. J. A. Haum, C. J. Paone, and R. C. Stirn, addendum 2 "Exposed Cores" supplement 2 to NEDO-10527, January 1973.

4. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 6, Amendment 12.

5. NEDC-31654P, "Maximum Extended Operating Domain Analysis for Brunswick Steam Electric Plant," February 1989.

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

The limiting values for APLHGR when conformance to the operating limit is performed by hand calculation are provided in the CORE OPERATING LIMITS REPORT for each fuel type and, when required, for the most limiting lattice for multiple lattice fuel bundle types. Power and flow dependent adjustments are provided in the CORE OPERATING LIMITS REPORT to assure that the fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from off-rated conditions.

This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10 CFR 50.46 and that the fuel design analysis limits specified in NEDE-24011-P-A (Reference 1) will not be exceeded.

Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice operating at the bounding power history, meet the fuel design limits specified in Reference 1. No single fuel rod follows, or is capable of following, this bounding power history. This bounding power history is used as the basis for the fuel design analysis APLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR 50 Appendix K to demonstrate that the permissible planar power (APLHGR) limits comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

The Technical Specification APLHGR limit is the most limiting composite of the fuel mechanical design analysis APLHGR and the ECCS APLHGR limit.

3/4.2.2 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.2 are derived from an established fuel cladding integrity Safety Limit MCPR approved by the NRC and an analysis of abnormal operational transients. 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).

Details on how evaluations are performed, on the methods used, and how the MCPR limit is adjusted for operation at less than rated power and flow conditions are given in References 1 and 2 and the CORE OPERATING LIMITS REPORT.

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

References:

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

2. NEDC-31654P, "Maximum Extended Operating Domain Analysis for Brunswick Steam Electric Plant," February 1989.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.5.1.

b. Seismic event analysis, Specification 4.3.5.1.2.

c. Accident Monitoring Instrumentation, Specification 3.3.5.3.

d. Fire detection instrumentation, Specification 3.3.5.7.

e. Reactor coolant specific activity analysis, Specification 3.4.5.

f. ECCS actuation, Specifications 3.5.3.1 and 3.5.3.2.

g. Fire suppression systems, Specifications 3.7.7.1, 3.7.7.2, 3.7.7.3, and 3.7.7.5.

h. Fire barrier penetration, Specification 3.7.8.

i. Liquid Effluents Dose, Specification 3.11.1.2.

j. Liquid Radwaste Treatment, Specification 3.11.1.3.

k. Dose - Noble Gases, Specification 3.11.2.2.

l. Dose - Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form, Specification 3.11.2.3.

m. Gaseous Radwaste Treatment, Specification 3.11.2.4.

n. Ventilation Exhaust Treatment, Specification 3.11.2.5.

o. Total Dose, Specification 3.11.4.

p. Monitoring Program, Specification 3.12.1.b.

q. Primary Containment Structural Integrity, Specification 4.6.1.4.2

CORE OPERATING LIMITS REPORT

6.9.3.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for Specification 3.2.1 including core flow and core power adjustments.

CORE OPERATING LIMITS REPORT (Continued)

b. The core flow and core power adjustments for Specification 3.2.2.1

c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.2.2.1 and 3.2.2.2.

d. The rod block monitor upscale trip setpoint and allowable value for Specification 3.3.4.

and shall be documented in the CORE OPERATING LIMITS REPORT.

6.9.3.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

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

b. The May 18, 1984 and October 22, 1984 NRC Safety Evaluation Reports for the Brunswick Reload Methodologies described in:

1. Topical Report NF-1583.01, "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors," February 1983.

2. Topical Report NF-1583.02, "Methods of RECORD," February 1983.

3. Topical Report NF-1583.03, "Methods of PRESTO-B," February 1983.

4. Topical Report NF-1583.04, "Verification of CP&L Reference BWR Thermal-Hydraulic Methods Using the FIBWR Code," May 1983.

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.10 RECORD RETENTION

Facility records shall be retained in accordance with ANSI-N45.2.9-1974.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY, et al.

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

DOCKET NO. 50-324

1.0 INTRODUCTION

By letter dated March 29, 1989 (Reference 1), Carolina Power & Light Company (the licensee), requested changes to the Brunswick Steam Electric Plant (BSEP) Technical Specifications (TS) to permit operation in the maximum extended operating domain (MEOD). The MEOD encompasses both a maximum extended load line limit (MELLL) and increased core flow (ICF) regions of the power/flow map. The licensee's submittal included proposed Figure and Table changes to the BSEP TS relating to average power range monitor (APRM), rod block settings and the limiting power/flow line.

Enclosed with the March 29, 1989 letter was a General Electric Company (GE) analysis of the consequences of operation in the MEOD (Reference 2) to justify the proposed changes. The requested changes are in the general categories identified as:

(1) deletion or modification of specifications having cycle-specific parameter limits and replacement of certain values of these limits with a reference to a core operating limits report (COLR) and deletion of the redundant linear heat generation limit from the Specifications;

(2) modification of the flow-biased APRM scram and rod block equations to accommodate the expanded operating domain; and

(3) modification of the rod block monitor (RBM) trip setpoints and RBM system requirements.

The proposed changes are addressed individually in the following Safety Evaluation (SE) Section 2.0. The evaluation includes reference to separate, related approvals of the specific areas regarding elimination of cycle-specific parameter values and deletion of the linear heat generation rate (Reference 3) and the loss-of-coolant accident analysis (Reference 4).

2.0 EVALUATION

2.1 Elimination of Flow-Biased APRM Scram and Rod Block Trip Setpoint Setdown Requirement

In the current BSEP Technical Specifications, the flow-biased APRM scram and rod block trip setpoints are reduced (setdown) when the core MAPLHGR is greater than the fraction of rated thermal power (FRTP). This requirement is associated with a now obsolete Hench-Levy minimum critical heat flux ratio criterion. The GE analysis (Reference 2) enclosed with the BSEP submittal includes the results from the analyses that were performed to determine a set of flow and power dependent fuel thermal limits minimum critical power ratio (MCPR) and MAPLHGR radio that would be needed to satisfy the pertinent licensing criteria if APRM setdown were eliminated. The new limits should (1) prevent violation of the MCPR safety limit, (2) keep the fuel thermal-mechanical performance within the design and licensing basis, and (3) keep peak cladding temperature and maximum cladding oxidation within allowable limits. The results of the analyses with approved analytical methods are as follows:

(1) New generic power-dependent MCPR and MAPLHGR limit adjustment factors are developed which consider two power ranges distinguished by a defined power (30 percent of rated) below which reactor scrams on turbine stop valve closure and turbine control valve fast closure are bypassed. The MAPLHGR relation is a factor, MAPFAC(P) which is multiplied by the rated MAPLHGR limit to obtain the power-dependent MAPLHGR limit.

(2) New generic flow-dependent MCPR and MAPLHGR limit adjustment factors are developed. These factors are determined from analyses of slow flow-runout transients with the requirement that the peak transient linear heat generation rate does not exceed the fuel design basis values. In connection with this amendment request, the new flow-dependent MCPR factors, MCPR(F), will replace the previously used MCPR multiplier, K_f, which will no longer be utilized.

The development of the adjustment factors described above used a TS improvement program, APRM/RBM Technical Specification (ARTS) Program, which, in part, supports both the implementation of updated fuel thermal limits and the elimination of the APRM trip setdown requirements. The transient analysis results discussed in the following SE Section 2.2 are a part of the ARTS program. The ARTS concept has been used over the past few years successfully and with staff approval by a number of utilities having BWR reactors. Its use in the present amendment request is acceptable to the staff.

2.2 Abnormal Operational Transients

All transients of Chapter 15 of the Brunswick FSAR were considered for the MEOD. The transients reevaluated were generator load reduction without turbine bypass (LRNBP), feedwater controller failure maximum demand (FWCF), and inadvertent high pressure coolant injection (HPCI) events. The potentially limiting LRNBP and FWCF events were evaluated at the power/flow conditions corresponding to the MELLL bounding point (100 percent power, 75 percent flow)

and the ICF bounding point (100 percent power, 105 percent flow). The HPCI event was analyzed at both points with an additional 2 percent power uncertainty allowance as prescribed by the use of the approved GEMINI methodology. Reference 2 presents the results of cycle-specific calculations, using standard methodology, for the reevaluated events at the bounding condition of ICF for BSEP Units 1 and 2. These are presented for ODYN options A and B and for each fuel type presently loaded in the BSEP units and are compared with the values for standard operating conditions. The results indicate that the LRNBP is limiting under ICF conditions for each unit. All transient analyses were done with approved methodology (Reference 5).

GE has also examined other events and affected system components related to the requested extensions. These include overpressure protection, LOCA events, pressure differentials and vibration response on reactor internals and fuel assemblies. The results show that design limits will not be exceeded. The containment loss-of-coolant accident (LOCA) response was analyzed and the results show no significant impact of the MEOD. A separate evaluation of the SAFER/GESTR LOCA methodology has been recently documented. The review of these various GE examinations has concluded that suitable analyses were performed and the results are comparable to other reviews and are acceptable for BSEP.

2.3 Modification of Flow-Biased APRM Scram and Rod Block Trip Equations

The MEOD proposal changes the APRM flux scram lines on the power/flow map and permits operation up to the new APRM flux scram line (0.66W + 64 percent) and up to the intersection with the 100 percent power line occurring at a flow of 75 percent. This is a standard change for MELLL. For ICF, the proposed flow increase is to 105 percent core flow at 100 percent power with a linear expansion to 110 percent core flow at 70 percent power. The increased flow would be allowed throughout a plant cycle. The flow-biased rod block trip equation is changed to 0.66W + 58 percent with a maximum value of 108 percent. The maximum value of 108 percent necessitates a modification to establish a clamping function for the rod block trip level (see Section 2.5).

2.4 Modification of RBM Trip Setpoints and RBM System Operability Requirements

The RBM system serves solely to mitigate the consequences of a rod withdrawal error (RWE) anticipated operational occurrence. A modified RBM system configuration is described in Reference 2 and will be implemented during the upcoming refueling outages for each BSEP unit. The process of defining RBM operating requirements entails a generic RWE analysis to determine that neither the safety limit nor the fuel thermal-hydraulic basis is jeopardized by the complete withdrawal of a single control rod. This modification is made in coordination with the ARTS program and allows the selection of the RBM setpoint such that the RWE analysis results are bounded by the limiting transient analysis (Section 2.2). The specific setpoints will be documented in a Core Operating Limits Report in accordance with approved procedures (Reference 3).

2.5 Plant Modifications

The aforementioned modifications to the APRM and RBM systems are necessary to ensure the availability of the expanded domain. This includes the modifications necessary to establish the clamping function associated with the flow-biased APRM rod block TS. These modifications are to be performed for BSEP Unit 2 during Refueling Outage 8 and BSEP Unit 1 during Refueling Outage 7. This SE may be considered applicable to both Units 1 and 2 since the basis for the changes is essentially the same. However, in accordance with the licensee's projected reload schedules, implementation of the specific TS changes on Unit 1 is not required until the next Unit 1 refueling outage.

2.6 Technical Specification Changes for MEOD

The proposed changes to the BSEP TS are identified in Enclosures 4 and 5 of the licensee's submittal. The changes include editorial changes to the TS index entries, deletion of the definitions of core MAPLHGR ratio and fraction of rated thermal power, a revision to the Bases Sections associated with the specific changes to the Safety Limits and Limiting Safety System Settings. The basis for the changes and the staff conclusions are detailed in the previous SE sections.

Changes to the Limiting Conditions for Operation (LCO) were also proposed as follows:

(1) TS 3.1.4.3. - Rod Block Monitor

Changes are necessary to identify the new APPLICABILITY ranges and ACTION statements associated with the proposed RBM system operability requirements (see Section 2.4).

(2) TS 3.2.1 - Average Planar Linear Heat Generation Rate

Changes are necessary to identify the new flow and power-dependent adjustment factors.

(3) TS 3.2.2.1 - Minimum Critical Power Ratio

Changes are necessary to identify the adjustment of the MCPR limit for core flow and power, to delete the APRM setpoint Specification, and to renumber the Specifications for continuity.

(4) TS 3.3.4 and footnote (f) to Table 3.3.4-1

Changes are necessary to identify the withdrawal ranges and ACTION statements associated with control rod block instrumentation similar to Item (1) above.

For changes (1) through (4) above, revised BASES discussion paragraphs were made to be compatible with the revised Specifications.

Changes were made to Section 6 (Administrative Controls) to state that the core operating limits associated with the requested changes will be identified in the Core Operating Limits Report.

The requested changes are the same for both BSEP units. The staff has reviewed the information submitted for operation of BSEP Units 1 and 2 with extended operating regions. Based on this review, the staff concludes that appropriate documentation was submitted to justify the proposed TS changes. The changes identified in the licensee submittal are acceptable as proposed.

3.0 ENVIRONMENTAL CONSIDERATIONS

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site; and that there should be no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the FEDERAL REGISTER (54 FR 32706) on August 9, 1989, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

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

Principal Contributor: M. McCoy

Dated: October 12, 1989

REFERENCES

1. Letter (NLS-89-060) from A. B. Cutter (CP&L) to Document Control Desk (USNRC) dated March 29, 1989 transmitting a Request for Licensing Amendment on Maximum Extended Operating Domain.

2. NEDC-31654P, "Maximum Extended Operating Domain Analysis for Brunswick Steam Electric Plant," February 1989 (GE Nuclear Energy) transmitted as Enclosure 7 to Reference 1.

3. Letter, E. G. Tourigny (USNRC) to L. W. Eury (CP&L) dated May 25, 1989. Subject: Issuance of Amendment No. 131 to Facility Operating License No. DPR-71 and Amendment No. 161 to Facility Operating License No. DPR-62 - Brunswick Steam Electric Plant, Units Nos. 1 and 2, Regarding Changes on Elimination of Cycle Dependent Parameter Values and Deletion of the Linear Heat Generation Rate Limit (TAC Nos. 66153 and 66154).

4. Letter, E. G. Tourigny (USNRC) to L. W. Eury (CP&L) dated June 1, 1989. Subject: SAFER/GESTR-LOCA Analysis, Brunswick Steam Electric Plant, Units 1 and 2 (TAC Nos. 72845/72855).

5. General Electric Standard Application for Reactor Fuel (NEDE-24011-P-A-US) with Safety Evaluation incorporated.

AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
E. Tourigny
N. Le
L. Spessard (MNBB 3701)
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
B. Grimes (9A2)
T. Meeks (4) (P1-137)
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