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Docket No. 50-325

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

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 29 to Facility Operating License NO. DPR-71 for the Brunswick Steam Electric Plant (BSEP) Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your applications dated May 23, May 30, and supplements dated June 4, and June 25, 1980.

The amendment changes the Technical Specifications to establish revised safety and operating limits for BSEP Unit 1 operation in operating Cycle No. 3. The amendment also changes the safety-relief valve pressure setpoints for 3 of the 11 valves to provide a minimum nominal lift setting differential for each valve pair of 20 psi.

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

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 29 to DPR-71
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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SEE PREVIOUS YELLOW FOR CONCURRENCE

| | | | | | | |
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| DATE | 6/25/80 | 6/26/80 | 6/ /80 | 6/ /80 | 6/ /80 | |

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

July 1, 1980

Docket No. 50-325

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

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Operating Reactors Branch #2
Division of Licensing

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2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

cc:

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Bolivia, North Carolina 28422

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Division of Policy Development
116 West Jones Street
Raleigh, North Carolina 27603

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

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

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

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

Mr. Fred Tollison
Plant Manager
P. O. Box 458
Southport, North Carolina 28461

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. DPR-71

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

   A. The applications for amendment by Carolina Power & Light Company (the licensee) dated May 23, May 30, as supplemented June 4, and June 25, 1980 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. 29, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 1, 1980

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ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages. The changed area of the revised page is reflected by a marginal line.

| Remove | Insert |
|---|---|
| III/IV | III/IV |
| V/VI | V/VI |
| 3/4 2-1/2 | 3/4 2-1/2 |
| 3/4 2-5/6 | 3/4 2-5/6 |
| 3/4 2-7/8 | 3/4 2-7/8 |
| 3/4 2-9/10 | 3/4 2-9/10 |
| - | 3/4 2-11 |
| 3/4 3-41/42 | 3/4 3-41/42 |
| B3/4 2-1/2 | B3/4 2-1/2 |
| B3/4 2-3/4 | B3/4 2-3/4 |
| 3/4 4-4 | 3/4 4-4 |

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## INDEX

## SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS



## BASES

# INDEX

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

#### LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR's) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 or 2.3.1-6.

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 or 3.2.1-6, initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

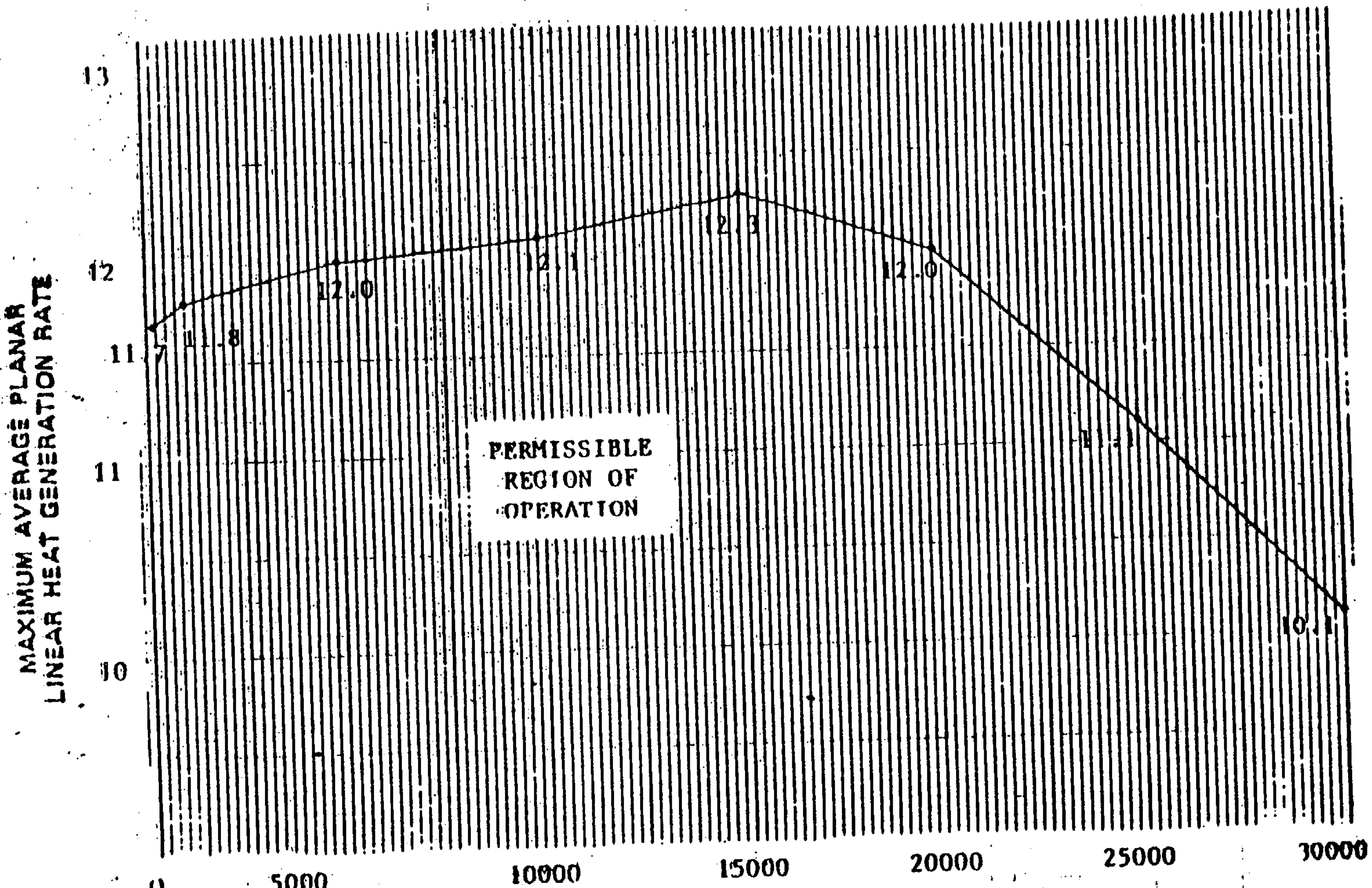
#### SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGR's shall be verified to be equal to or less than the applicable limit determined from Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 or 3.2.1-6:

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

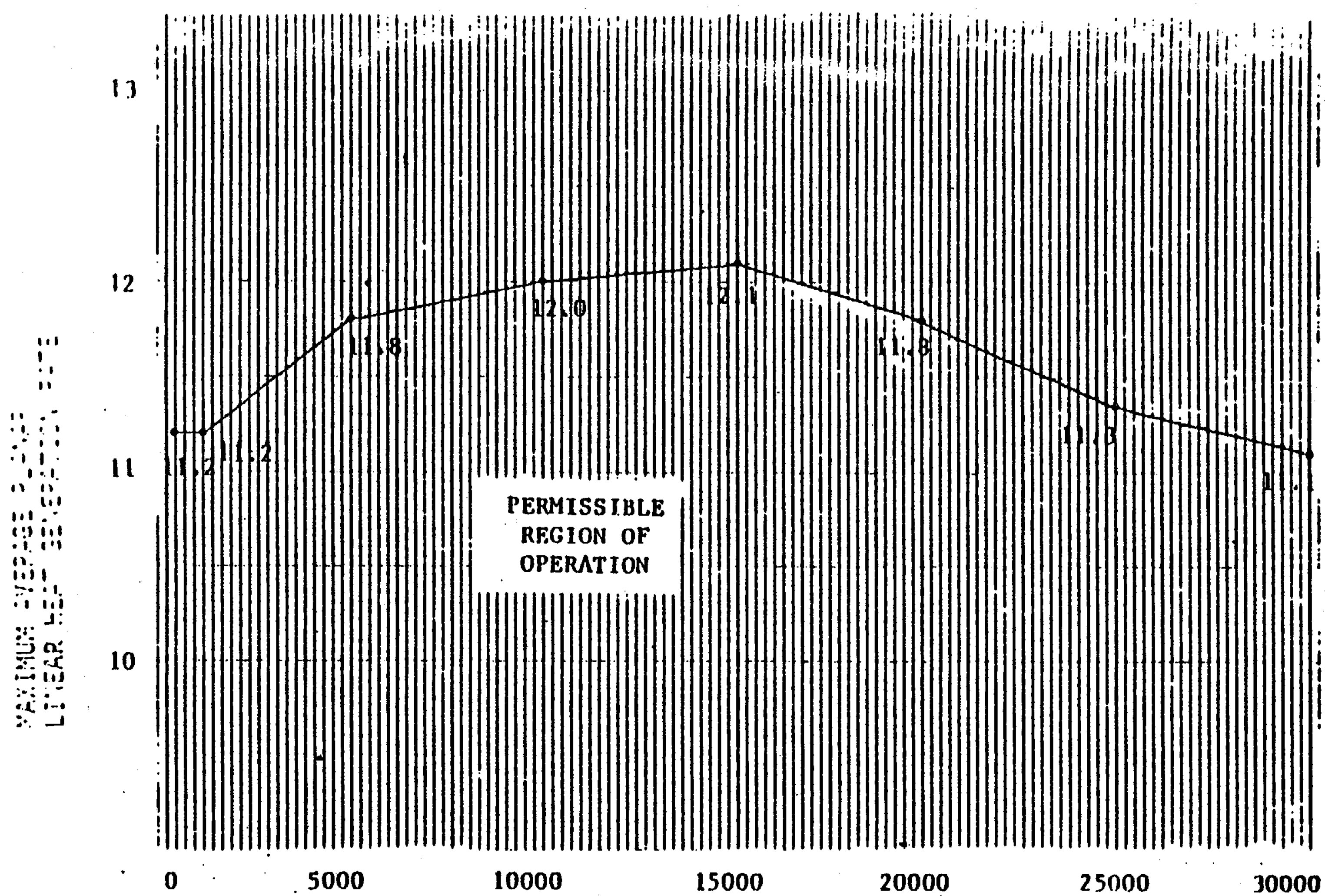
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.




AVERAGE PLANAR EXPOSURE (MWd/t)
FUEL TYPE 1 (8 x 8)

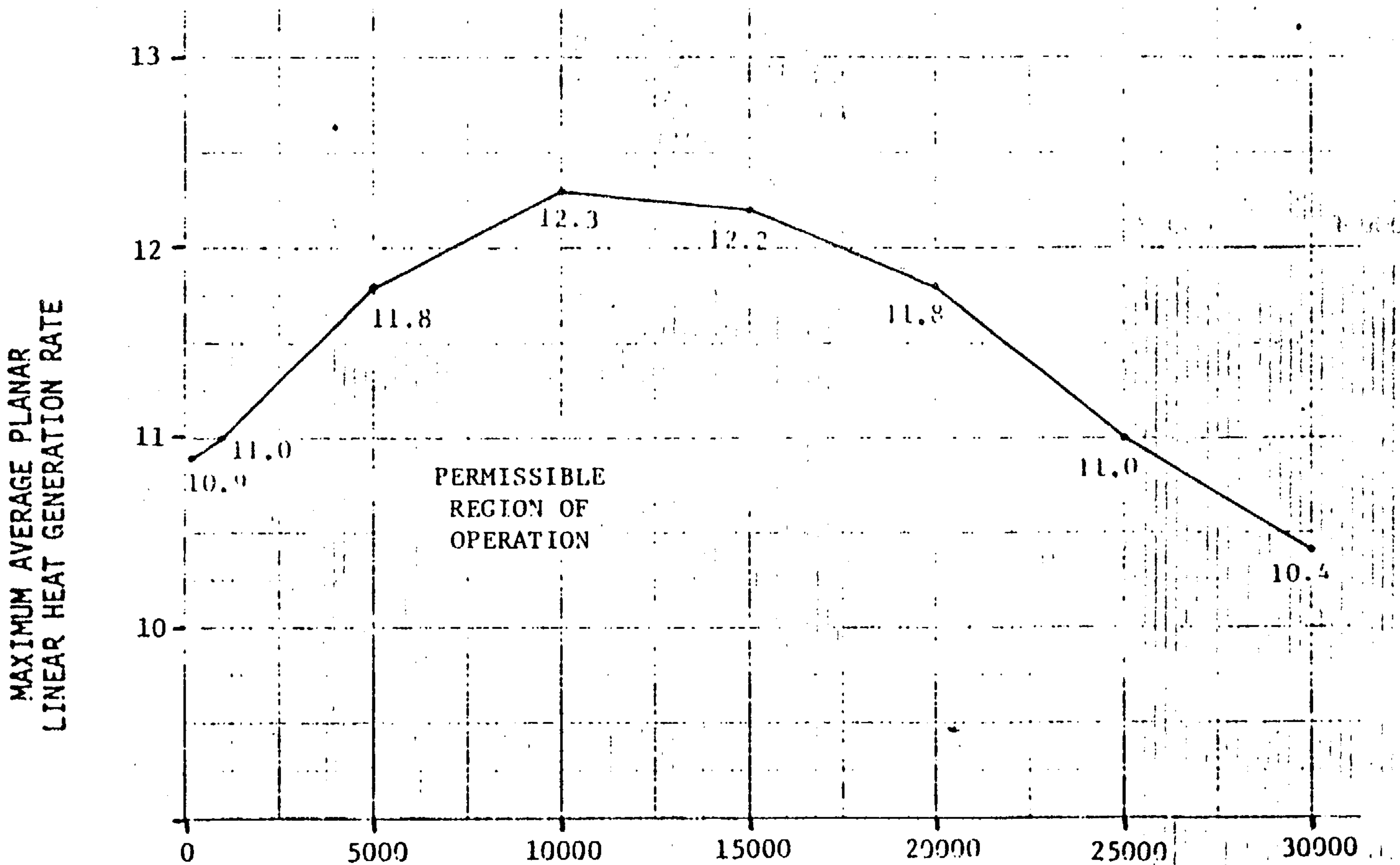
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

Figure 3.2.1-1



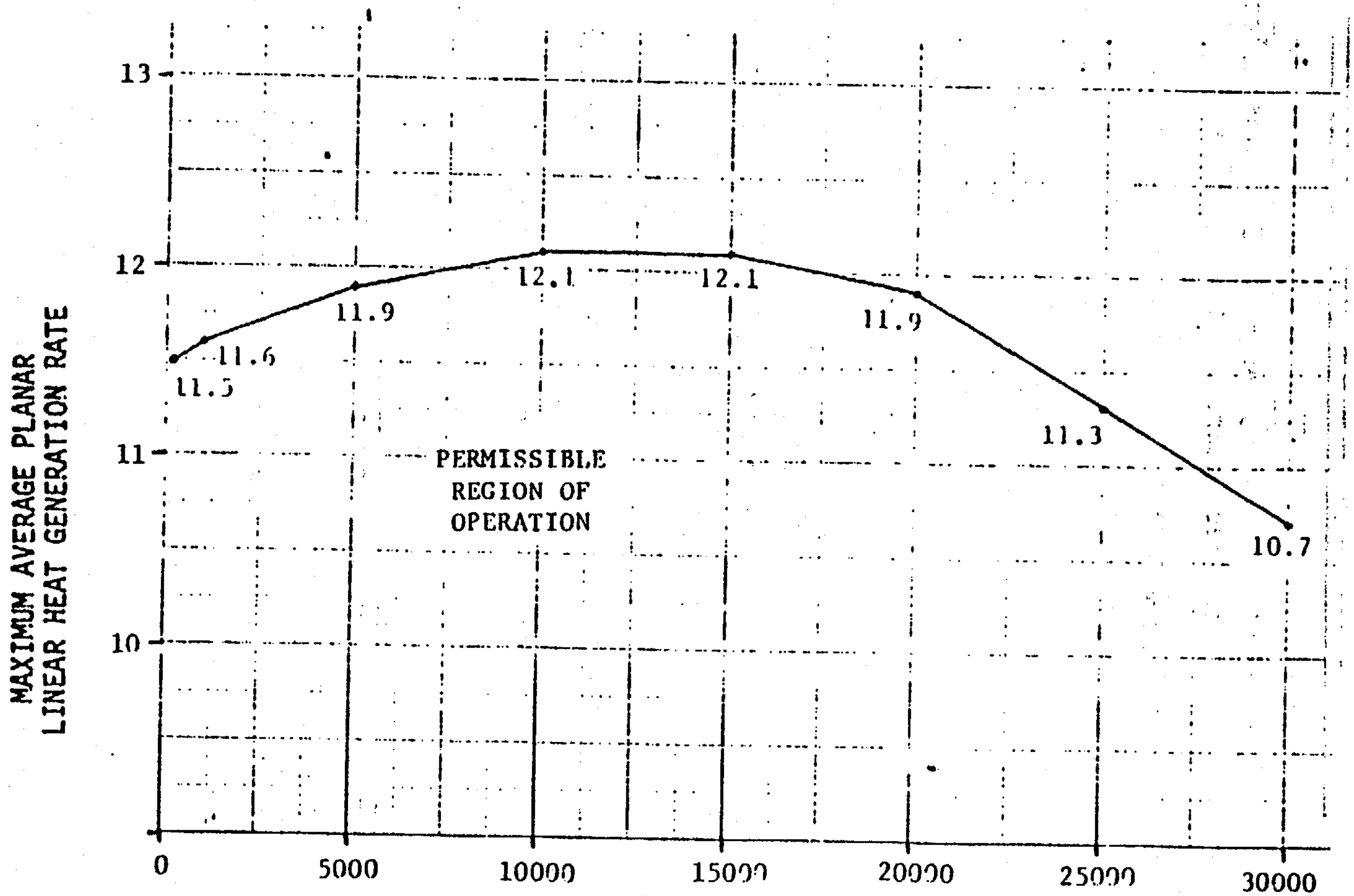

PLANAR AVERAGE EXPOSURE (MWd/t)
FUEL TYPE 8DRB283 (8x8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-4




PLANAR AVERAGE EXPOSURE (MWd/t)
FUEL TYPE P8DRB285 (P8x8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-5




PLANAR AVERAGE EXPOSURE (MWd/t)
FUEL TYPE P8DRB285 (P8x8R)
MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-6

## 3/4.2.2 APRM SETPOINTS

### LIMITING CONDITION FOR OPERATION

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

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

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

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

Design TPF for: 8 x 8 fuel = 2.45.
8 x 8R fuel = 2.48.
P8 x 8R fuel = 2.48.

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

ACTION:

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

### SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

b. 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 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, shall be equal to or greater than MCPR x the $K_f$ shown in Figure 3.2.3-1 where MCPR values are:

| | BOC3* to EOC3** -2000 MWD/t | EOC3-2000 MWD/t to EOC3 |
|---|---|---|
| 8x8 fuel | 1.24 | 1.30 |
| 8x8R fuel | 1.24 | 1.30 |
| P8x8R fuel | 1.30 | 1.32 |

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 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 continue corrective action so that MCPR is equal to or greater than the applicable limit within 4 hours or reduce THERMAL POWER TO LESS THAN 25% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

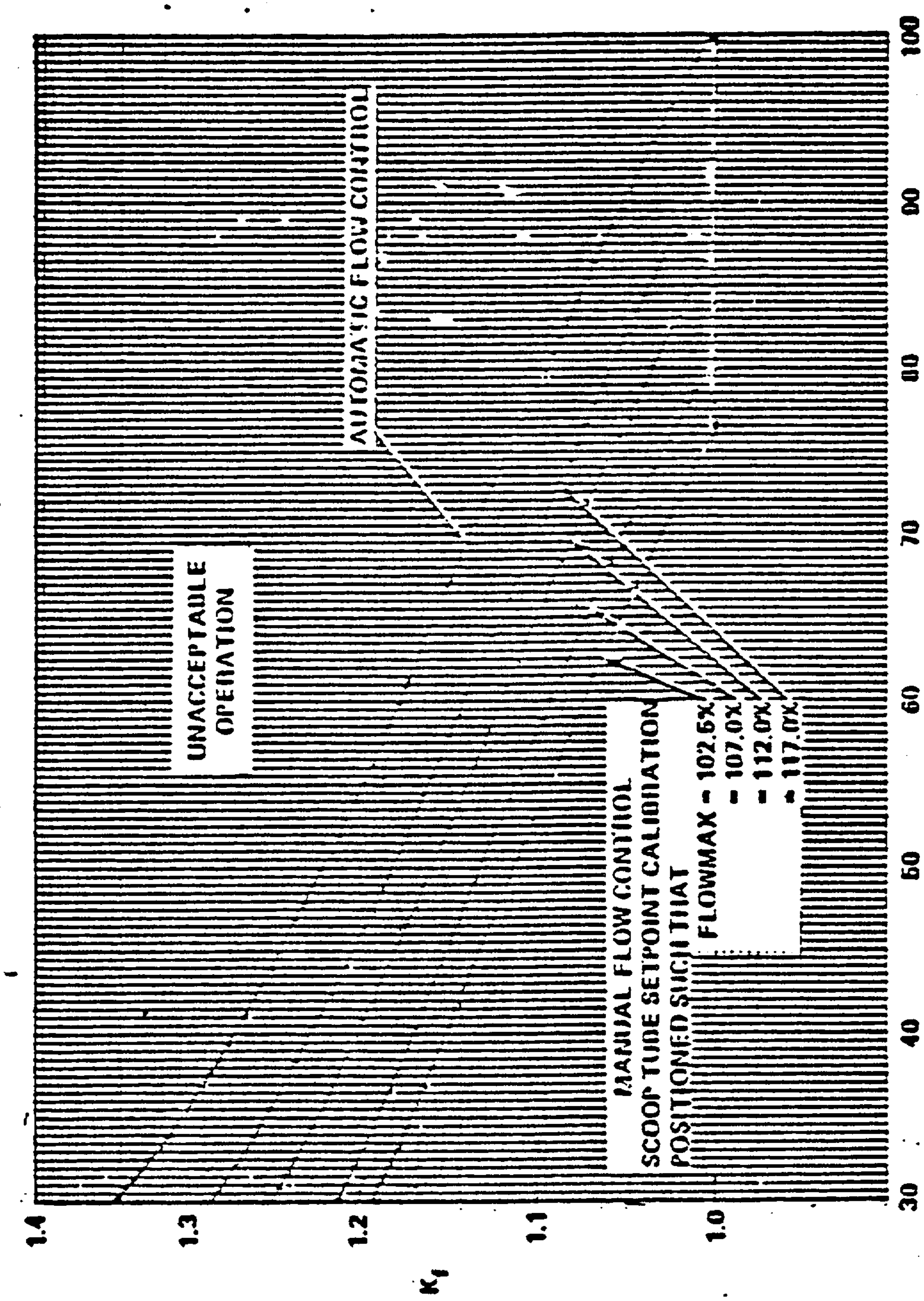
4.2.3 MCPR, as a function of core flow, shall be determined to be equal to or greater than the applicable limit determined from Figure 3.2.3-1:

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 with a LIMITING CONTROL ROD PATTERN for MCPR.

*Beginning of Cycle 3.
**End of Cycle 3.




CORE FLOW (%)

$K_1$ FACTOR

FIGURE 3.2.3-1

## 3/4.2.4 LINEAR HEAT GENERATION RATE

### LIMITING CONDITION FOR OPERATION

3.2.4 All LINEAR HEAT GENERATION RATES (LHGR's), shall not exceed 13.4 kw/ft.

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

ACTION:

With the LHGR of any fuel rod exceeding 13.4 kw/ft., 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 LHGRs shall be determined to be equal to or less than 13.4 kw/ft:

a. At least once per 24 hours,

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

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

TABLE 3.3.4-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

NOTE

* When THERMAL POWER exceeds the preset power level of the RWM and RSCS.

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

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

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

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

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

## TABLE 3.3.4-2

## CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

| TRIP FUNCTION AND INSTRUMENT NUMBER | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. APRM (C51-APRM-CH.A,B,C,D,E,F) | | |
| a. Upscale (Flow Biased) | $\le (0.66\ W + 42\%)\ \frac{T^*}{MTPF}$ | $\le (0.66\ W + 42\%)\ \frac{T^*}{MTPF}$ |
| 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 (C51-RBM-CH.A,B) | | |
| a. Upscale | $\le (0.66W + 41\%)\ \frac{T^*}{MTPF}$ | $\le (0.66\ W + 41\%)\ \frac{T^*}{MTPF}$ |
| b. Inoperative | NA | NA |
| c. Downscale | ≥ 3/125 of full scale | ≥ 3/125 of full scale |
| 3. SOURCE RANGE MONITORS (C51-SRM-K600A,B,C,D) | | |
| a. Detector not full in | NA | NA |
| b. Upscale | $\le 1 \times 10^5$ cps | $\le 1 \times 10^5$ cps |
| c. Inoperative | NA | NA |
| d. Downscale | ≥ 3 cps | ≥ 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 | ≤ 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 |

*T=2.43 for 8 x 8 fuel.
T=2.48 for 8 x 8 R fuel.
T=2.48 for P8x8R fuel.

## 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 a assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification APHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 and 3.2.1-6.

The calculational procedure used to establish the APLHGR shown on Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6 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 analyses assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6, (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limitation as described in Reference 2, are included in the reflooding calculations.

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

## Bases Table B 3.2.1-1

## SIGNIFICANT INPUTS 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 ..............$10.96 \times 10^6$ 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** |
|---|---|---|---|---|
| All | 8 x 8 | 13.4 | 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.

## 3/4.2.2 APRM SETPOINTS

The fuel cladding integrity safety limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of 2.45 for 8 x 8 fuel and 2.48 for 8 x 8R and P8 x 8R fuel. The scram setting and rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.0 in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and peak flux indicates a TOTAL PEAKING FACTOR greater than 2.45 for 8 x 8 fuel and 2.48 for 8 x 8R and P8 x 8R fuel. The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.

## 3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and an analysis of abnormal operational transients[1]. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting as given in Specification 2.2.1.

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

The limiting transient which determines the required steady state MCPR limit is the turbine trip with failure of the turbine by pass. This transient yields the largest $\Delta$ MCPR. When added to the Safety Limit MCPR of 1.07 the required minimum operating limit MCPR of Specification 3.2.3 is obtained. Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4.4 of NEDO-20360[4] and on core parameters shown in Reference 3, response to Items 2 and 9.

MINIMUM CRITICAL POWER RATIO (Continued)

The evaluation of a given transient begins with the system initial parameters shown in Attachment 5 of Reference 6 that are input to a GE-core dynamic behavior transient computer program described in NEDO-10802(5). Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDO-20566(1). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $K_f$ factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the $K_f$ factor. Specifically, the $K_f$ factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the $K_f$ factors assure that the operating limit MCPR of Specification 3.2.3 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the $K_f$ factors assure that the Safety Limit MCPR will not be violated should the most limiting transient occur at less than rated flow.

The $K_f$ factor values shown in Figure 3.2.3-1 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The $K_f$ factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the $K_f$ factors were calculated such that the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the $K_f$.

## 3/4.4.2 SAFETY/RELIEF VALVES

### LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of all reactor coolant system safety/relief valves shall be OPERABLE with lift settings within ± 1% of the following values.*#

4 Safety-relief valves @ 1105 psig.
4 Safety-relief valves @ 1115 psig.
3 Safety-relief valves @ 1125 psig.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety valve function of one safety/relief valve inoperable, restore the inoperable safety valve function of the valve to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the safety valve function of two safety/relief valves inoperable, restore the inoperable safety valve function of at least one of the valves to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the safety valve function of more than two safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

### SURVEILLANCE REQUIREMENTS

4.4.2 The safety valve function of each of the above required safety/relief valves shall be demonstrated OPERABLE by verifying that the bellows on the safety/relief valves have integrity, by instrumentation indication, at least once per 24 hours.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperature and pressure.

*From Spring, 1980 until the maintenance outage in Sept., 1980, the safety-relief valve lift settings shall be arranged such that each safety-relief valve pair has a minimum nominal lift setting differential of 20 psi and shall be within ± 1% of the following values:

2 Safety-relief valves @ 1095 psig
3 Safety-relief valves @ 1105 psig
3 Safety-relief valves @ 1115 psig
3 Safety-relief valves @ 1125 psig

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 29 TO FACILITY LICENSE NO. DPR-71

## CAROLINA POWER & LIGHT COMPANY

## BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

## DOCKET NO. 50-325

A. Brunswick Steam Electric Plant, Unit No. 1, Operating Cycle No. 3 - Reload Application

By letter dated May 23, 1980, Carolina Power and Light Company (CP&L or licensee) requested revisions to the Technical Specifications to complete the second refueling of Brunswick Steam Electric Plant, Unit No. 1 (BSEP) and begin Cycle 3 operation.

The staff was assisted in the Safety Evaluation of the BSEP 1 reload licensing analysis by our technical consultant, Brookhaven National Laboratory (BNL). The following evaluation was submitted by BNL on June 19, 1980.

I. INTRODUCTION

In a recent letter[1] to the NRC Carolina Power and Light (CP&L) Company has requested revisions to the Technical Specifications for its Brunswick Steam Electric Plant (BSEP) Unit No. 1, and submitted General Electric's (GE) "Supplemental Reload Licensing Submittal for BSEP Unit 1 Reload 2".[2]

The above documents containing plant specific data, along with GE's BWR generic reload document[3] and NRC's Safety Evaluation Report[4] (SER) on the generic reload document have been reviewed. Additional bundle data describing basic nuclear characteristics[5] of one of the new bundle types used in the BSEP-1 Reload-2 core, recently submitted by GE, have also been reviewed.

This report presents a summary of our safety evaluation based on our review of the above documents.

CP&L's BSEP 1 is a BWR-4 plant. The Cycle 3 core is expected to contain 560 8 x 8 bundles including 156 fresh assemblies. These fresh assemblies are of the prepressurized retrofit type and would constitute 28% of the core.

Our evaluation of the BSEP 1 Reload 2 is limited to the items discussed in the following sections. Our acceptance of the results discussed in these sections is strictly limited to the criteria set forth by the USNRC in USNRC's own SER's referred to in this report. Brookhaven National Laboratory (BNL), acting as technical consultants to the USNRC, has not performed independent analyses to verify either the methods or the results and accuracy of the GE analyses. To establish acceptance of the results of GE's calculations, BNL has relied on NRC's SERs.

# 2. EVALUATION

## 2.1 Nuclear Characteristics

There are two types of fresh bundles planned for reload in the Brunswick 1 Cycle 3 core: 16 Reload 2 bundles designated as P8DRB265H and 140 Reload 2 bundles labelled P8DRB285. Reference 2 lists the types and numbers of the previously irradiated fuel assemblies. Figure 1 of Reference 2 shows the reference core loading pattern. We note that in near-central locations as well as near the periphery there are four-bundle control cells in which two out of the four assemblies are fresh. The beginning of cycle (BOC) cold eigenvalue with the strongest control rod fully withdrawn and all other rods fully inserted is reported to be 0.972. Technical Specifications require that adequate cold shutdown margin be demonstrated at BOC-3 with the highest worth rod withdrawn. Results shown in sections 4 and 5 indicate that both the control rod system and the standby liquid control system will have adequate shutdown margins under the most reactive conditions of the core.

## 2.2 Thermal Hydraulics

### 2.2.1 Fuel Cladding Integrity Safety Limit

The calculated safety limit MCPR of 1.07 for BWR reload cores such as Brunswick 1 Reload 2 has been found to be acceptable for the 8 x 8R (Reference 4) and P8 x 8R (Reference 5) fuels. This safety limit implies that during a transient characterized by an MCPR of 1.07, 99.9% of the fuel rods in the core are expected to avoid boiling transition.

### 2.2.2 Operating Limit MCPR (OLMCPR)

To insure that the fuel cladding integrity safety limit is not violated during any abnormal operational transient, the most limiting transients have been re-analyzed for Brunswick 1 Reload 2. The OLMCPR is obtained by adding to the safety limit the maximum CPR value for the most limiting transient for each fuel type. The OLMCPR values for the 8 x 8, 8 x 8R and P8 x 8R fuel types are given for the two exposure ranges in Section II of Reference 2.

2.2.2.1 Transient Analysis Methods

The methods employed for the transient calculations have been described in Reference 3. NRC approval of these methods has been documented in Reference 4. Inputs and initial condition parameters for the transient analysis calculations are given in the tables of Sections 6 and 7 of Reference 2. NRC's evaluation of the methods used to generate these reload-unique values is also included in Reference 4.

2.2.2.2 Transient Analysis Results

Transient events analyzed were the generator load rejection without bypass, feedwater controller failure, loss of 100°F feedwater heating and control rod withdrawal error. Reload-unique initial conditions and transient input parameters were assumed to be those listed in Sections 6 and 7 of Reference 2. Results of these analyses are listed in Sections 9 and 10. We have not verified independently the results of these analyses. However, the differences between these results and those of Brunswick 2 are small and consistent with the two designs. Also, as mentioned in Section 2.2.2.1 above, the generic methods employed in carrying out the calculations[3] have received approval by the NRC.[4]

## 2.3 Accident Analysis

### 2.3.1 ECCS Appendix K Analysis

In a supplement[4] to the earlier Safety Evaluation Report of GE's Licensing Topical Report of the Generic Reload Application,[3] application of the ECCS-LOCA (Appendix K) models used in the 8 x 8 retrofit reload fuel which was found to be "generically acceptable" has been extended to cover the P8 x 8R fuel. Based on that SER,[4] the proposed MAPLHGR limits for the prepressurized 8 x 8 retrofit fuel are found to be acceptable.

### 2.3.2 Control Rod Drop Accident

Results of the control rod drop accident analysis are shown in Figures 9 through 13 of Reference 2. These figures are intended to demonstrate that the curves plotted are appropriately covered by the bounding analysis. The latter is based on the assumption that the reactivity excursion caused by the rod drop will not result in a fuel enthalpy greater than 280 cal/gm at any axial fuel location in any fuel rod. The methods[3] used in carrying out these analyses have been approved by the NRC (Section 7.3 of Reference 4). We find these results to be acceptable.

### 2.3.3 Fuel Loading Error

Using the NRC approved methodology for the analysis of misoriented and misloaded bundles,[3] the GE Supplemental Reload Licensing document[2] reports that in the limiting event which results from a rotated P8 x 8R bundle, there is adequate margin to insure no loss in fuel integrity. We thus find the results of this analysis to be acceptable.

### 2.3.4 Overpressure Analysis

The NRC has determined that the effects of fuel prepressurization are well accounted for in vessel overpressurization analyses.[4] Accordingly, we agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure.

### 2.5 Technical Specifications

The Technical Specifications have been changed to include specifications associated with the new, prepressurized type bundles as well as the corresponding surveillance requirements, regarding the Average Planar Linear Heat Generation Rates (APLHGR's), the APRM and Rod Block Monitor setpoints. These Technical Specifications changes reflecting the introduction of the new type of bundles have been reviewed and found acceptable.

### 2.6 Densification Power Spiking

It is acceptable to remove the 8 x 8, 8 x 8R and P8 x 8R spiking penalty factor from the Technical Specification of those BWR's for which it can be demonstrated that the predicted worst case maximum transient LHGR's, when augmented by the power spike penalty, do not violate the exposure-dependent safety limit LHGR's. The Brunswick plant meets the above criterion. Section 10, Rod Withdrawal Error and Appendix E Linear Heat Generation Rate for Bundle Loading Error, of Reference 2 include the densification effect in the reported LHGR value for all 8 x 8 type assemblies. On the basis of these data, we find that the Licensee meets the requirements on the densification power spiking.

### 2.7 Thermal Power Monitor

Operation of Brunswick 1 Cycle 3 with the Thermal Power Monitor (TPM) feature is acceptable provided the USNRC has already approved this option in the previous cycle for this plant. It was agreed in a recent conference call[6] that CP&L will provide the USNRC with the details of the earlier TPM approval for this plant.

## 2.8 Startup Plans

In the supplemental submittal[2], no mention is made of a startup test program for Brunswick 1.

We were informed[6] that CP&L plans to follow at Brunswick 1 the same startup test plans as those detailed in an earlier letter regarding the startup of the last two cycles of Brunswick 2. We received a verbal commitment from CP&L that the latter will inform the USNRC by letter on the startup test plans for the new cycle.

## References


1. Letter, E. E. Utley (Carolina Power and Light Company) to T. A. Ippolito (USNRC) May 23, 1980 and Revised Technical Specifications for Brunswick Steam Electric Plant Unit 1.

2. "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 1 Reload 2," NEDO-24239, General Electric Company, January 1980.

3. "General Electric Boiling Water Reactor Generic Reload Application," NEDE-24011-P-A.

4. Letter, T. A. Ippolito (USNRC) to R. Gridley (GE) dated April 16, 1979 and enclosed SER.

5. Letter, R. E. Engel (GE) to T. A. Ippolito (USNRC), "Specific Bundle Enrichment Nuclear Characteristics," April 30, 1980.

6. Telephone Conference Call, CP&L, GE, USNRC and BNL, June 19, 1980.

## B. Brunswick Steam Electric Plant, Unit No. 1, Safety Relief Valve Setpoints

### 1.0 Introduction

By letter dated May 30, 1980, the licensee requested a temporary change in the setpoint values of 3 of the 11 BESP-1 safety-relief valves. This change was necessitated by the postponement of a major Mark I Containment modification effort until the fall of 1980. The modification involved the installation of T-quenchers in the torus to replace the existing paired discharge line design. System reserve requirements for the late summer of 1980 forced deferment of the planned Mark I Containment modification program. CP&L has previously committed to install the T-quencher modification for both Brunswick Units in the spring of 1980. (Letter dated January 30, 1980.)

### 2.0 Discussion

During a visual inspection of inaccessible snubbers performed in December 1979, damaged snubbers were found on the safety relief valve F013H tailpipe. It is believed the damage occurred following a reactor scram on November 20, 1979 when safety relief valves (SRV's) F013F, G, and H automatically lifted. SRV's F013F and H share one of the 5 paired discharge headers in the torus. The eleventh SRV (F013K) discharges directly into the torus through a single header. SRV's F013F and H had a setpoint differential pressure spread of 10 psi. Subsequent analysis indicated that the damage may have been caused by a water slug in the exhaust line of the paired discharge header.

The Mark I torus modifications will rearrange the SRV exhaust lines in the torus such that each valve will have a separate T-quencher. By eliminating the shared discharge headers, the likelihood for future tailpipe damage is reduced.

In lieu of the T-quencher modification, the licensee is proposing to increase the setpoint differential pressure spread for each of the paired SRV's to 20 psi. Since there have been no cases of simultaneous or near-simultaneous liftings of SRV pairs with a 20 psi setpoint differential, the licensee feels that this change will provide adequate assurance of SRV tailpipe integrity until the Mark I T-quencher modifications are installed in fall 1980.

### 3.0 Evaluation

To determine the adequacy of the proposed SRV setpoint change, we reviewed the staff's SER for BSEP Units 1 and 2 Supplement No. 2 dated December 23, 1974; Amendment No. 31 to DPR-62 dated October 6, 1977; and Amendment No. 14 to DPR-71 dated September 11, 1978.

3.0 Transient Analysis Methods

In a recent Safety Evaluation* the staff concluded that the 8x8R GEXL correlation used by GE in the reload analysis for non-equilibrium cores has conservatisms which are equivalent to the 7x7 and 8x8 GEXL correlations previously approved by the staff. However, the data supporting the application of GEXL to 8x8R fuel have never been submitted for staff review in accordance with established procedures. We will require that this data base be submitted so that the staff can complete its review and that this issue be formally resolved prior to operation in future cycles.

For future cycles also, the REDY code will not be acceptable for use in calculating core response to pressurization transients. Reference NRC letter to G. G. Sherwood (GE) from Dick Denise dated January 23, 1980.

4.0 Conclusion

By letter dated June 25, 1980, CP&L confirmed that the Thermal Power Monitor feature previously approved for BSEP-1 will be used this operating cycle. By the same letter, CP&L confirmed that the startup physics test program previously approved and followed for the previous BSEP-1 cycle will be used for this operating cycle also.

Based on our review of the consultant's Safety Evaluation and the CP&L letter of June 25, 1980, we find the proposed operation in cycle 3 to be acceptable.

Dated: July 1, 1980

*Amendment No. 62 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station, Dated May 20, 1980.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-325

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 29 to Facility Operating License No. DPR-71, issued to Carolina Power & Light Company (the licensee) for operation of the Brunswick Steam Electric Plant, Unit No. 1 (the facility), located in Brunswick County, North Carolina. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to establish revised safety and operating limits for BSEP Unit 1 operation in operating Cycle No. 3. The amendment also changes the safety-relief valve pressure setpoints for 3 of the 11 valves to provide a minimum nominal lift setting differential for each valve pair of 20 psi.

The applications for amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of the amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(5) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the applications for amendment dated May 23, May 30, as supplemented June 4, and June 25, 1980, (2) Amendment No. 29 to License No. DPR-71, and (3) the Commission's related Safety Evaluation. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. 20555, and at the Southport Brunswick County Library, 109 West Moore Street, Southport, North Carolina 28461. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 1st day of July 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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