DISTRIBUTION:
Docket
NRC PDR
Local PDR
ORB-1 Reading
VStello
KRGoller
ASchwencer
CTrammell
SSheppard
Attorney, OELD
DEisenhut
RBaer
JCarter
JMcGough
JRBuchanan
V. Rooney

TBAbernathy
ACRS (16)
CMiles, OPA
DRoss
BHarless
BJones (4)
BScharf (15)

Docket No. 50-324

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 38 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your requests dated August 3, August 22, and September 22, as supplemented on November 10 and 21, 1977.

The amendment changes the Technical Specifications for the facility to establish revised safety and operating limits for operation in Cycle 2 with both 7x7 and new 8x8 fuel, and includes changes resulting from a reevaluation of Emergency Core Cooling System (ECCS) cooling performance submitted by CP&L on September 22, 1977, in compliance with the Commission's Order for Modification of License dated March 11, 1977. This reevaluation corrected the errors identified in the March 11, 1977 Order and included the effect of other recently approved model changes in the ECCS evaluation models. The CP&L submittal of September 22, 1977, therefore satisfies the action required by the March 11, 1977 Order, and no further action by CP&L with respect to this Order is required.

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

Sincerely,

/s/

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 38 to License No. DPR-62
2. Safety Evaluation
3. Notice

subject to Note change in to C. Trammell 11/23/77

Const. 1

GD

| OFFICE | DOR:ORB-1 | DOR:ORB-1 | DOR:RSB | OELD | DOR:ORB-1 | DOR:AD/ORs |
|---|---|---|---|---|---|---|
| SURNAME | SSheppard | CTrammell:esp | RBaer | | ASchwencer | KRGoller |
| DATE | 11/21/77 | 11/21/77 | 11/21/77 | 11/23/77 | 11/23/77 | 11/ /77 |

NRC FORM 318 (9-76) NRCM 0240

U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

George F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

John J. Burney, Jr., Esquire
Burney, Burney, Sperry & Barefoot
110 North Fifth Avenue
Wilmington, North Carolina 28401

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

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

Office of Intergovernmental
Relations
116 West Jones Street
Raleigh, North Carolina 27603

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

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

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE➤ | | | | | | |
| SURNAME➤ | | | | | | |
| DATE➤ | | | | | | |

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

Docket No. 50-324

November 23, 1977

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 38 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your requests dated August 3, August 22, and September 22, as supplemented on November 10 and 21, 1977.

The amendment changes the Technical Specifications for the facility to establish revised safety and operating limits for operation in Cycle 2 with both 7x7 and new 8x8 fuel, and includes changes resulting from a reevaluation of Emergency Core Cooling System (ECCS) cooling performance submitted by CP&L on September 22, 1977, in compliance with the Commission's Order for Modification of License dated March 11, 1977. This reevaluation corrected the errors identified in the March 11, 1977 Order and included the effect of other recently approved model changes in the ECCS evaluation models. The CP&L submittal of September 22, 1977, therefore satisfies the action required by the March 11, 1977 Order, and no further action by CP&L with respect to this Order is required.

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

Sincerely,

for A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 38 to
   License No. DPR-62
2. Safety Evaluation
3. Notice

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

George F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

John J. Burney, Jr., Esquire
Burney, Burney, Sperry & Barefoot
110 North Fifth Avenue
Wilmington, North Carolina 28401

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

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

Office of Intergovernmental
Relations
116 West Jones Street
Raleigh, North Carolina 27603

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

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

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

# CAROLINA POWER AND LIGHT COMPANY

## DOCKET NO. 50-324

## BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

## AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. DPR-62

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

   A. The applications for amendment by Carolina Power & Light Company (the licensee) dated August 3, August 22 and September 22, as supplemented on November 10 and 21, 1977, 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 applications, 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 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. 38 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Revise Appendix A Technical Specifications as follows:

1. Remove Figures 2.1-1
   2.1-2
   3.1-2C

2. Remove the following pages and replace with identically numbered revised pages:

   1.1-1
   1.1-2
   3.1-1, 3.1-2
   3.1-2A
   3.1-3
   3.2-2
   3.2-24
   3.2-29
   3.2-41
   3.2-42 , 3.6-8, 3.6-26
   5.0-1
   Figure 3.1-1
   Figure 3.1-2A
   Figure 3.1-2B

3. Add the following new pages:
   Figures 3.2-1a through e.

Marginal lines indicate changes.

SAFETY LIMIT

## 1.1 Fuel Cladding Integrity

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. When the reactor pressure is equal to or greater than 800 psia or core flow $\geq$ 10%, the minimum critical power ratio shall be $\geq$ 1.06.

LIMITING SAFETY SYSTEM SETTINGS

## 2.1 Fuel Cladding Integrity

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded:

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM - The APRM scram trip setpoint (S) shall be:

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

where:

S = Setting in percent of rated power (2436 MWt)

W = Recirculation loop flow in percent of design

T = Lowest value of the ratio of design TPF divided by the MTPF obtained for any type of fuel in the core ($T \leq 1.0$).

Design TPF for 8x8 fuel = 2.45
Design TPF for 7x7 fuel = 2.60

| SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTINGS |
|---|---|

1.1 Fuel Cladding Integrity (Cont'd)

B. When the reactor pressure is less than 800 psia, or core cooling flow is less than 10 percent of design, the reactor thermal power shall not exceed 25% rated power.

To ensure that the safety limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its primary source signal. The safety limit shall be assumed to be exceeded when scram is accomplished by a means other than the primary source signal.

2.1.A Neutron Flux Scram (Cont'd)

2. APRM - When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.

3. IRM - the IRM flux scram setting shall be $\leq$ 120/125 of scale.

B. APRM Control Rod Block

The APRM Rod Block trip set point ($S_{RB}$) shall be:

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

The definitions used above for the APRM scram trip apply.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

## 3.1 Reactor Protection System

Applicability:

Applies to the operability of plant instrumentation and control systems required for reactor safety.

Objective:

To specify the limits imposed on plant operation by those instrument and control systems required for reactor safety.

Specification:

A. Plant Operation

Plant operation at any power level shall be permitted only in accordance with Table 3.1-1.

B. System Response

The designated system response time from actuation of the sensor contact or trip output to the de-energization of the scram solenoid relay shall not exceed 100 milliseconds.

C. Minimum Critical Power Ration (MCPR)

During steady-state power operation (25% or greater), MCPR as a function of core flow shall be equal to or greater than MCPR x the $K_f$ shown in Figure 3.1-1, where:

MCPR (7x7) = 1.26
MCPR (8x8) = 1.30

## 4.1 Reactor Protection System

Applicability:

Applies to the surveillance of the plant instrumentation and control systems required for reactor safety.

Objective:

To specify the type and frequency of surveillance to be applied to those instrument and control systems required for reactor safety.

Specification:

A. Plant Operation

Instrumentation systems shall be functionally tested and calibrated as indicated in Table 4.1-1.

B. System Response

The system response time will be checked prior to initial fuel loading.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Protection System (Cont'd)

D. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation*, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.2-1a through e.

*25% power or higher.

E. Local Linear Heat Generation Rate (LHGR)

During steady-state power operation (25% power or greater), all linear heat generation rates (LHGRs) as a function of core height for any fuel rod in an assembly shall not exceed the applicable maximum LHGR shown in Figures 3.1-2a and b.

SURVEILLANCE REQUIREMENTS

4.1 Reactor Protection System (Cont'd)

D. Average Planar Linear Heat Generation Rate (APLHGR)

The maximum ratio of the limiting value for APLHGR as a function of average planar exposure to the APLHGR value (APLHGR RATIO) for each type of fuel shall be determined daily during reactor power operation at ≥ 25% rated thermal power.

E. Local Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| | **4.1 Reactor Protection System (Cont'd)** |

F. Heat Flux and Maximum Total Peaking Factor

Once a day during reactor power operation and at constant power ≥ 25% the maximum peak heat flux and the total peaking factor shall be checked and the SCRAM and APRM Rod Block settings given by Specifications 2.1.A.1 and 2.1.B shall be calculated if the peaking factor exceeds the design values given in 2.1.A.1.

G. Inoperable Channels

When an instrument channel monitoring any variable in the reactor protection system (RPS) fails, its associated RPS trip system must be manually tripped if the minimum number of operable instrument channels per trip system cannot be met.

The failed instrument channel may be bypassed to permit functional testing of the untripped RPS trip system providing that the remaining operable instrument channels monitoring the same variable in the tripped trip system are functionally tested immediately prior to bypassing the inoperable instrument channel.

In no case shall the inoperable instrument channel be bypassed for greater than eight hours per each functional test of the untripped trip system.

Amendment No. 38

USEP 1 & 2

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

| Trip Function | Trip Settings | Modes in Which Functions Must be Operable Refuel (1) | Startup | Run | Min. No. Operable Instrument Channels Per Trip System (2) | Required Conditions When Minimum Conditions for Operation Are Not Satisfied (3) |
|---|---|---|---|---|---|---|
| 1. Mode switch C72A-S1 | | X | X | X | 1 | A |
| 2. Manual trip C72A-S3A, B | | X | X | X | 1 | A |
| 3. IRM High flux | ≤120/125 of scale | X | X | (12) | 3 | A |
| Inoperative | | X | X | (12) | 3 | A |
| 4. APRM High Flux (4,14) (flow bias) | ≤(0.66W+54) T* | | | X | 2 | B |
| (fixed)(14) | ≤120% of | | | X | 2 | B |
| Inoperative | | X | | X | 2(5) | B |
| Downscale | ≥3/125 of scale | | | X(13) | 2 | B |
| Startup | ≤15% of rated power | X | X | | 2 | A |
| 5. High reactor pressure B21-PS-N023A, B,C,D | ≤1045 psig | X(10) | X | X | 2 | A |
| 6. High drywell pressure C72-PS-N002A, B,C,D | ≤2 psig | X(11) | X(11) | X | 2 | A |
| 7. Reactor low water level #1 B21-LIS-N017A, B,C,D | ≥12.5 inch (6) | X | X | X | 2 | A |

*T is defined in Specification 2.1.A.1.



Amendment No.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.2.C. Control Rod Block Actuation | 4.2.C. Control Rod Block Actuation |
| 1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-11. | 1. Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2-11.<br><br>System logic shall be functionally tested as indicated in Table 4.2-11. |
| 2. The minimum number of operable instrument channels specified in Table 3.2-11 for the rod block monitor may be reduced by one for maintenance and/or testing provided that this condition does not last longer than 24 hours. | |
| 3. If one channel of the rod block monitor has been inoperable for more than 24 hours, control rod withdrawal shall be blocked; or the operating power level shall be limited such that the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod. | |
| D. Radiation Monitoring Systems - Isolation & Initiation Functions | D. Radiation Monitoring Systems - Isolation & Initiation Functions |
| 1. The limiting conditions for operation for Reactor Building ventilation system isolation and standby gas treatment system are given in Table 3.2-12. | 1. Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2-12.<br><br>System logic shall be functionally tested as indicated in Table 4.2-12. |
| E. Drywell Leak Detection | E. Drywell Leak Detection |
| The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2-13. | Instrumentation shall be calibrated and checked as indicated in Table 4.2-13. |
| F. Post Accident Monitoring Instrumentation | F. Post Accident Monitoring Instrumentation |
| 1. The limiting conditions for the instrumentation that provides surveillance information readouts are given in Table 3.2-14. | 1. Instrumentation shall be calibrated and checked as indicated in Table 4.2-14. |



Amendment No. 38

Amendment No

Amendment No. 38

BSEP-1 & 2

TABLE 3.2-7

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

CORE SPRAY SYSTEM - A & B (1)

| | Trip Function | Trip Level Setting | Minimum Number of Operable Instrument Channels per Trip System | Required Action When Minimum Conditions for Operation are not Satisfied | Remarks |
|---|---|---|---|---|---|
| 1. | High drywell pressure E11-PS-N011A,B,C,D | ≤ 2 psig | 2 | (2) | Initiates core spray and has contacts in LPCI, HPCI, ADS and diesel start. |
| 2. | Reactor low water level #3 B21-LIS-N031A,B,C,D | ≥ 17" above TAF (-147.5" indicated) | 2 | (2) | Initiates core spray and has contacts in LPCI, ADS and diesel start |
| 3. | Low reactor pressure B21-PS-N021A,B,C,D | 410 psig ± 15 psig | 2 | (2) | Permissive for opening core spray admission valves |
| 4. | Core spray pump start time relays E21-K16A,B | $14 < t < 16$ sec | 1 | (3) | Conjunctionally initiates sequential starting of CSCS pumps |
| 5. | Core spray pump discharge pressure interlock E21-PS-N008A,B E21-PS-N009A,B | 100 (± 10) psig | 2 | (3) | Prevents ADS actuation pending confirmation of core spray pump running interlock |

Amendment No. 38

BSEP-1 & 2

TABLE 3.2-8

INSTRUMENTATION THAT INITIATES OR CONTROLS
THE CORE AND CONTAINMENT COOLING SYSTEMS

LOW PRESSURE COOLANT INJECTION SYSTEM A & B (1)

| | Trip Function | Trip Level Setting | Minimum Number of Operable Instrument Channels per Trip System | Required Action When Minimum Conditions for Operation are not Satisfied | Remarks |
|---|---|---|---|---|---|
| 1. | High Drywell pressure E11-PS-N011A,B,C,D | ≤ 2 psig | 2 | (2) | Initiates LPCI and has contacts in core spray, HPCI, ADS, and diesel start |
| 2. | Reactor low water level #3 B21-LIS-N031A,B,C,D | ≥ 17" above TAF (-147.5" instrument) | 2 | (2) | Initiates LPCI and has contacts in core spray, ADS, and diesel start |
| 3. | Low reactor pressure | 410 psig ± 15 psig (LPCI injection) 310 psig ± 15 psig (recirculation discharge valve) | 2 | (2) | Permissive for opening LPCI injection valve, closing permissive for recirculation discharge valves |
| 4. | LPCI pump start time E21-K2A,B | 9 ≤ t ≤ 11 sec | 1 | (3) | In conjunction with loss of power initiate sequential starting of CSCS pumps |
| 5. | LPCI pump discharge pressure interlock E11-PS-N020A,B,C,D | 100 (± 9) psig | 2 | (3) | Prevents ADS actuation pending confirmation of LPCI or CS pump running interlock |

TABLE 3.2-11 (Cont'd)

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

| Trip Function | Minimum Number of Operable Instrument Channels (2) | Modes in Which Function Must Be Operable: Refuel | Startup | Run | Trip Setting | Remarks |
|---|---|---|---|---|---|---|
| c. Detector not in "full in" position, channels A through H, Relays C51-K9E through H, & J through M | 6 | X | X | | Detector motor module limit switch LS-4 not closed (detector not full in) | Bypassed in run mode. |
| d. Downscale IRM channels A through H, Relay C51-K51 | 6 | X | X | | ≥ 3/125 of Scale | Bypassed in run mode and when IRM is in RANGE 1. |
| 3. Average power range monitor | | | | | | |
| a. Upscale APRM channels A through F, Relays K1 & K7 | 4 | | | X | ≤ (0.66W+42) T* | |
| b. Inoperative APRM channels A through F Relays K2 & K8 | 4 | X | X | X | (1) | |
| c. Downscale APRM channels A through F Relays K3 & K9 | 4 | | | X | ≥ 3/125 of Full Scale | Only active when mode switch is in RUN |
| d. Upscale startup APRM channels A through F Relay K18 | 4 | X | X | | ≤ 12% power | Bypassed when in run mode. |

*T is defined in Specification 2.1.A.1.

TABLE 3.2 (Cont'd)

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

| Trip Function | Minimum Number of Operable Instrument Channels (3) | Modes in Which Function Must Be Operable Refuel | Startup | Run | Trip Setting | Remarks |
|---|---|---|---|---|---|---|
| 4. Rod block monitor | | | | | $\leq$ (0.66W+40) T | (5) |
| a. Upscale RBM channels A,B Relay K1 | 2 | | | X (4) | | |
| b. Downscale RBM channels A,B Relay K2 | 2 | | | X (4) | $\geq$ 3/125 of full scale | |
| c. Inoperative RBM channels A,B Relay K3 | 2 | | | X (4) | (1) | |

NOTES:

1) The inoperative trips are produced by the following conditions:

(a) SRM and IRM

1) Mode switch not in OPERATE
2) High voltage power supply voltage low
3) Circuit boards not in circuit

(b) APRM

1) Mode switch not in OPERATE
2) Less than 11 LPRM inputs
3) Circuit boards not in circuit

(c) RBM

1) Mode switch not in OPERATE
2) Circuit boards not in circuit
3) RBM fails to null
4) Less than required number of LPRM inputs for rod selected.

(2) If the minimum number of channels cannot be met for one out of two trip systems, seven days are allowed before requiring the affected trip system to be tripped. If both trip systems do not meet the minimum number of operable channels for operation, both trip systems shall be tripped.

(3) If the minimum number of channels per trip system cannot be met, see Specifications 3.2.C and 3.3.B.5 for required action.

(4) Only required operable when mode switch is in RUN and reactor power is $\geq$30%.

(5) T is defined in Specification 2.1.A.1.

## LIMITING CONDITIONS FOR OPERATION

3.6.F **Jet Pump Flow Mismatch**

Following 1-pump operation, the discharge valve of the low-speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

3.6.G **Structural Integrity**

The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

3.6.H **Condensate Demineralizers**

1. Regeneration of a condensate demineralizing resin charge shall occur before the predicted unused capacity of the resin reaches a minimum value of 30 pounds as chloride ions. Predicted capacity is based on resin salt splitting capacity, integrated flow or flow rate and influent conductivity.

2. At least one condensate demineralizer influent conductivity instrument shall be operable.

3.6.I **Natural Circulation**

Steady state operation with both recirculation pumps not operating is permitted for up to 12 hours. If both recirculation pumps are not restored to operation within 12 hours, the reactor shall be shutdown within the next 12 hours.

## SURVEILLANCE REQUIREMENTS

4.6.F **Jet Pump Flow Mismatch**

Following 1-pump operation, observe the speed of the faster pump to be less than 50% of its rated speed prior to opening the discharge valve of the lower speed pump.

4.6.G **Structural Integrity**

The nondestructive inspections listed in Table 4.6-1 shall be performed as specified. The results obtained from compliance with this Specification will be evaluated after five years and the conclusions of this evaluation will be reviewed with the AEC.

4.6.H **Condensate Demineralizers**

1. The percent of the remaining ion exchanger capacity of the anion resins shall be calculated and logged

   a. weekly when the influent conductivity is less than 0.3 umho/cm

   b. daily when the influent conductivity is equal to greater than 0.3 umho/cm.

**BASES:**

3.6.G and 4.6.G Structural Integrity (Cont'd)

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical and added sensitivity is required. Ultrasonic testing and/or radiography shall be used where defects can occur on concealed surfaces.

H. Condensate Demineralizers

The criteria of the resin monitoring program and the resin replacement program have been established to protect the reactor from high chloride level should a seawater leak occur in the main condenser. The criteria will provide for a minimum unused capacity of 30 pounds of chloride ion (50 percent depletion in a resin which is approaching 0.75 meq/ml) before a planned regeneration of a resin. Should a seawater leak occur when a resin has 30 pounds of capacity remaining; this criteria will allow a sufficient buffer for an orderly shutdown.

The resin depletion can be calculated using measured salt-splitting capacity, the flow through the bed, and the average influent conductivity. Based on this result, a depletion can be calculated which will assure a 30-pound chloride ion exchange reserve. Regeneration prior to this level of depletion will assure a sufficient ion exchange reserve for removal of chloride from the condensate system.

These factors form the basis for the frequency of sampling, analyzing, calculation, and logging surveillance requirements. The calculation and logging will be increased from a weekly basis to a daily basis when and if influent conductivity reaches 0.3 umho/cm or greater.

I. Natural Circulation

This specification limits the time that the reactor can operate in the natural circulation mode (both recirculation pumps not running). The natural circulation mode is the least stable mode of flow control.

Amendment No. 38

## 5.0 Major Design Features

### 5.1 Site Features

The Brunswick Steam Electric Plant is located in the southeastern portion of North Carolina in Brunswick County, approximately 135 miles SSE of Raleigh, North Carolina, 175 miles due east of Columbia, South Carolina and 150 miles NE of Charleston, South Carolina. The site is 16 miles south of the nearest boundary of Wilmington, North Carolina, in adjacent New Hanover County, and 2-1/2 miles north of Southport. Approximate coordinates of the Reactor Buildings are latitude 33°57.5'N and longitude 78°00.5'W. The site region is influenced by the Atlantic Ocean, which bounds the southern edge of Brunswick County, and the Cape Fear River, along the eastern border. The site is approximately five miles west and north of the Atlantic Ocean. Elevations range from sea level to about +30 feet mean sea level (MSL).

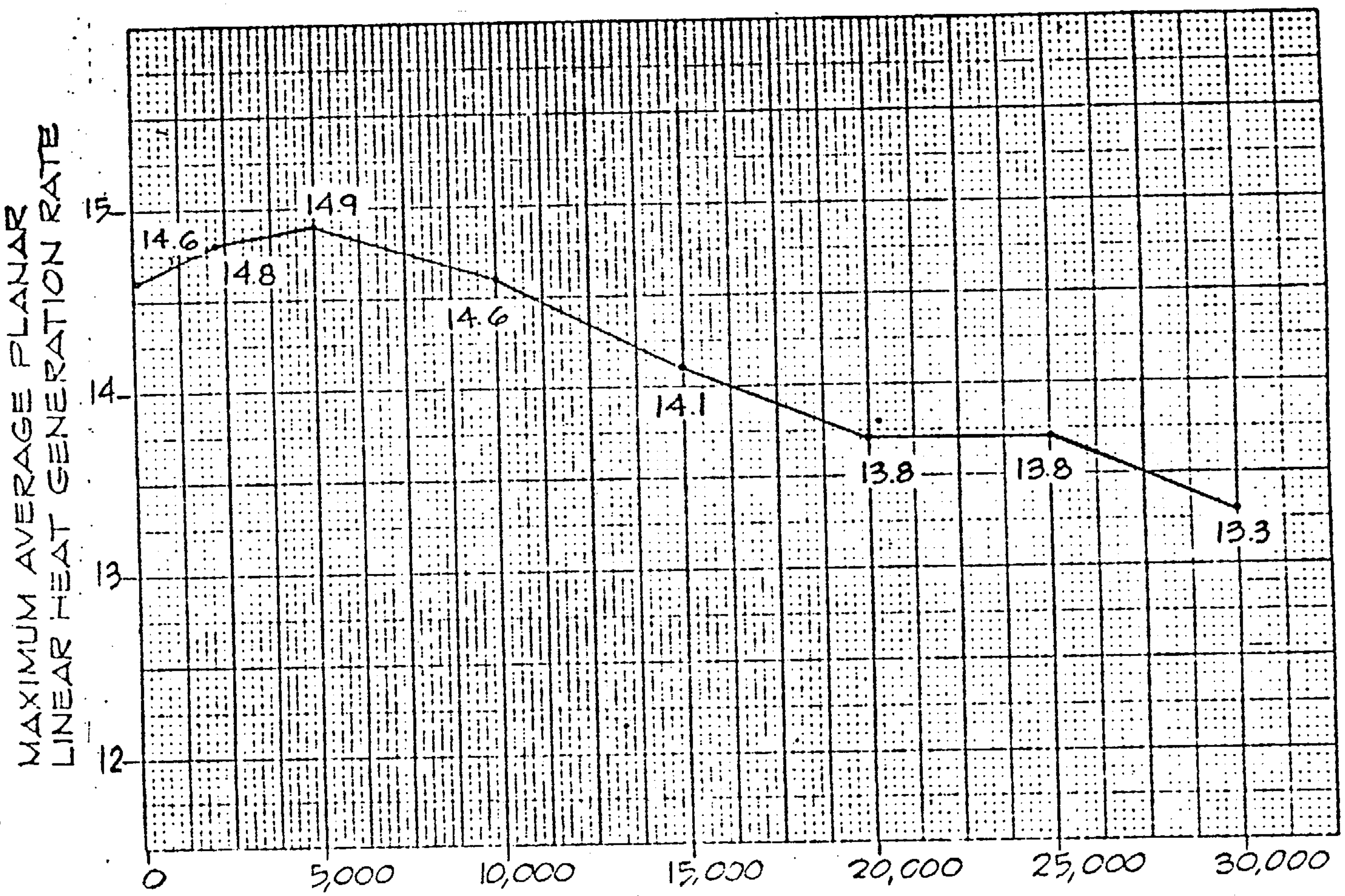
### 5.2 Reactor

A. The reactor core shall contain 560 fuel assemblies with each fuel assembly containing either 49 or 63 fuel rods clad with (Zircaloy 2). Each fuel rod shall have a nominal active fuel length of 144 or 146 inches and contain a maximum total weight of 4430 grams of $UO_2$. The initial core loading shall have a maximum enrichment of 2.47 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.8 weight percent U-235.

B. The reactor core shall contain 137 cruciform-shaped control rods. The control material shall be boron carbide powder ($B_4C$) compacted to approximately 70 percent of theoretical density.

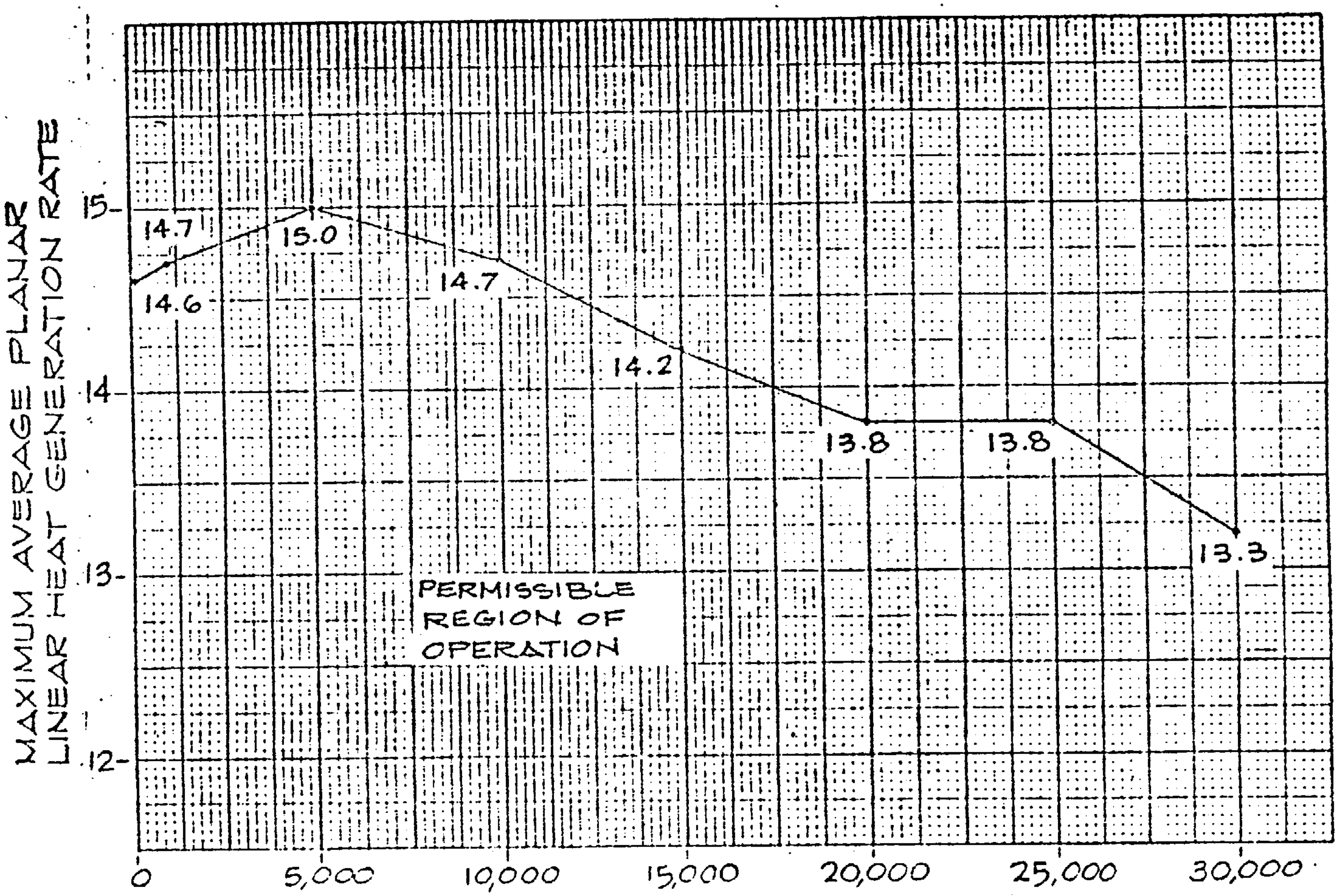
### 5.3 Reactor Vessel

The reactor vessel shall be as described in FSAR Table 4.2-2. The applicable design codes shall be as described in FSAR Section 4 and materials as described in FSAR Table 4.2-1.



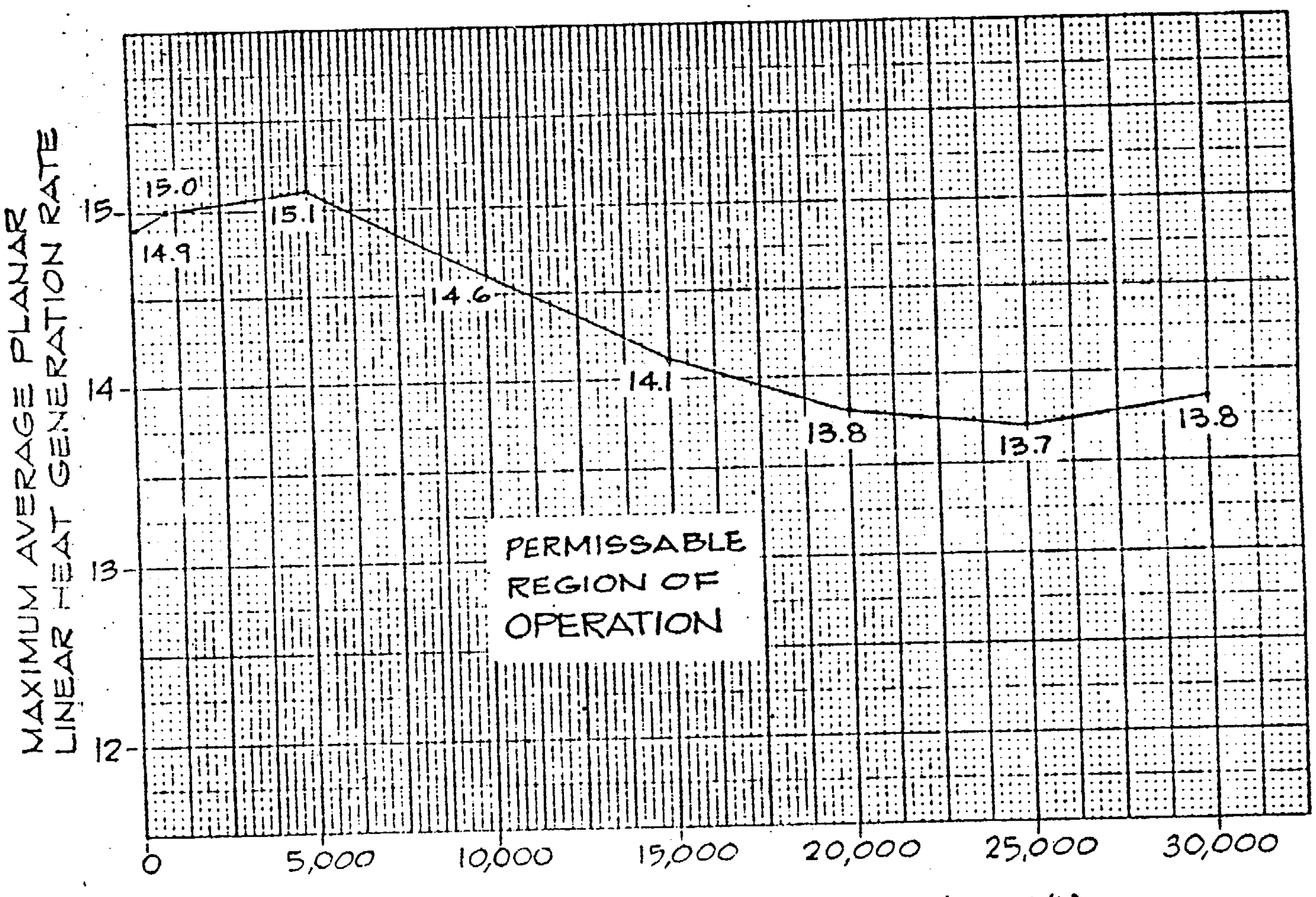

PLANAR AVERAGE EXPOSURE (MWd/t)
FUEL TYPE 1 & 2 (7 X 7)

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE
FIGURE 3.2-1a



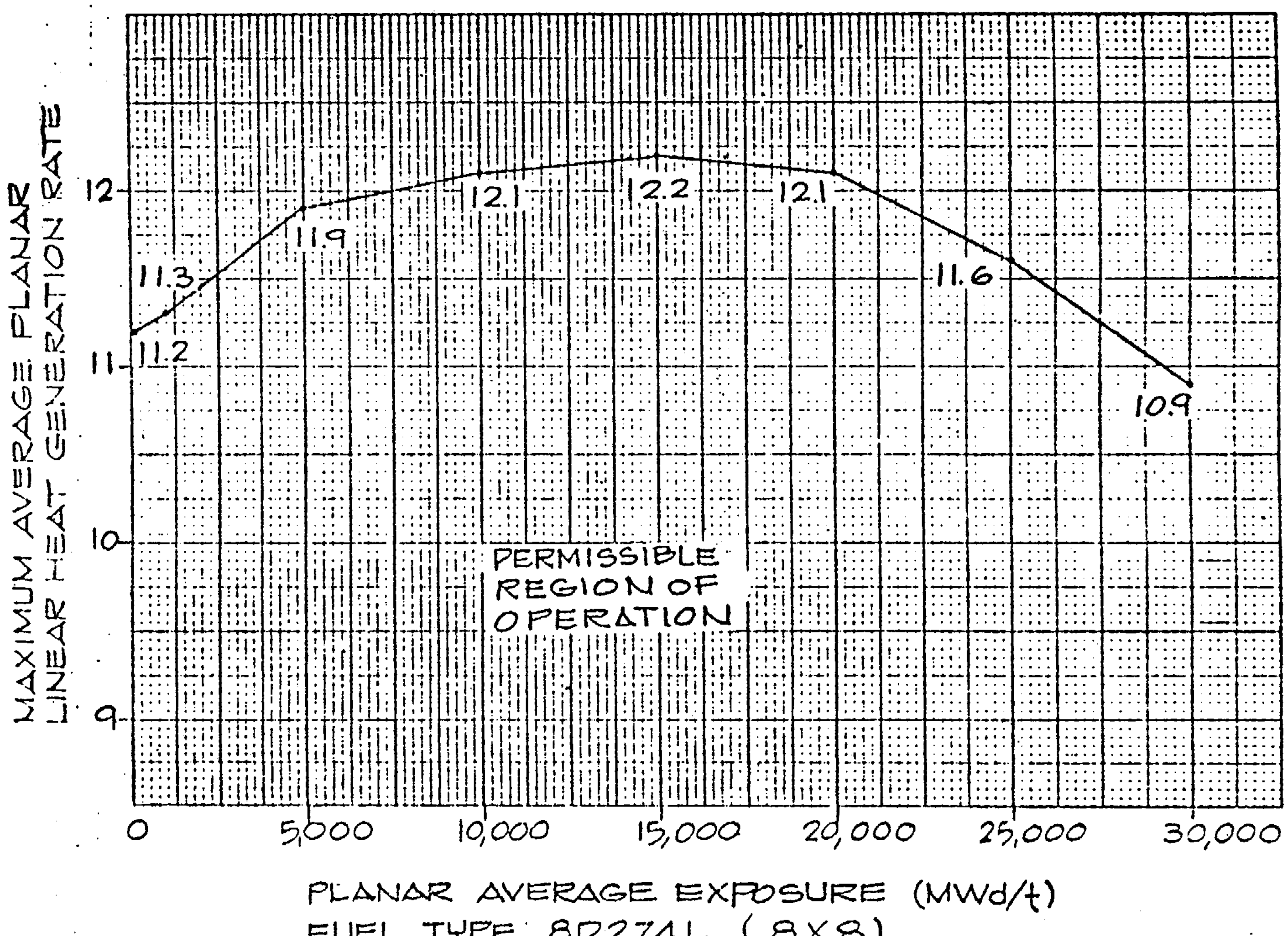

PLANAR AVERAGE EXPOSURE (MWd/t)
FUEL TYPE 3 (7X7)

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE
FIGURE 3.2-1b

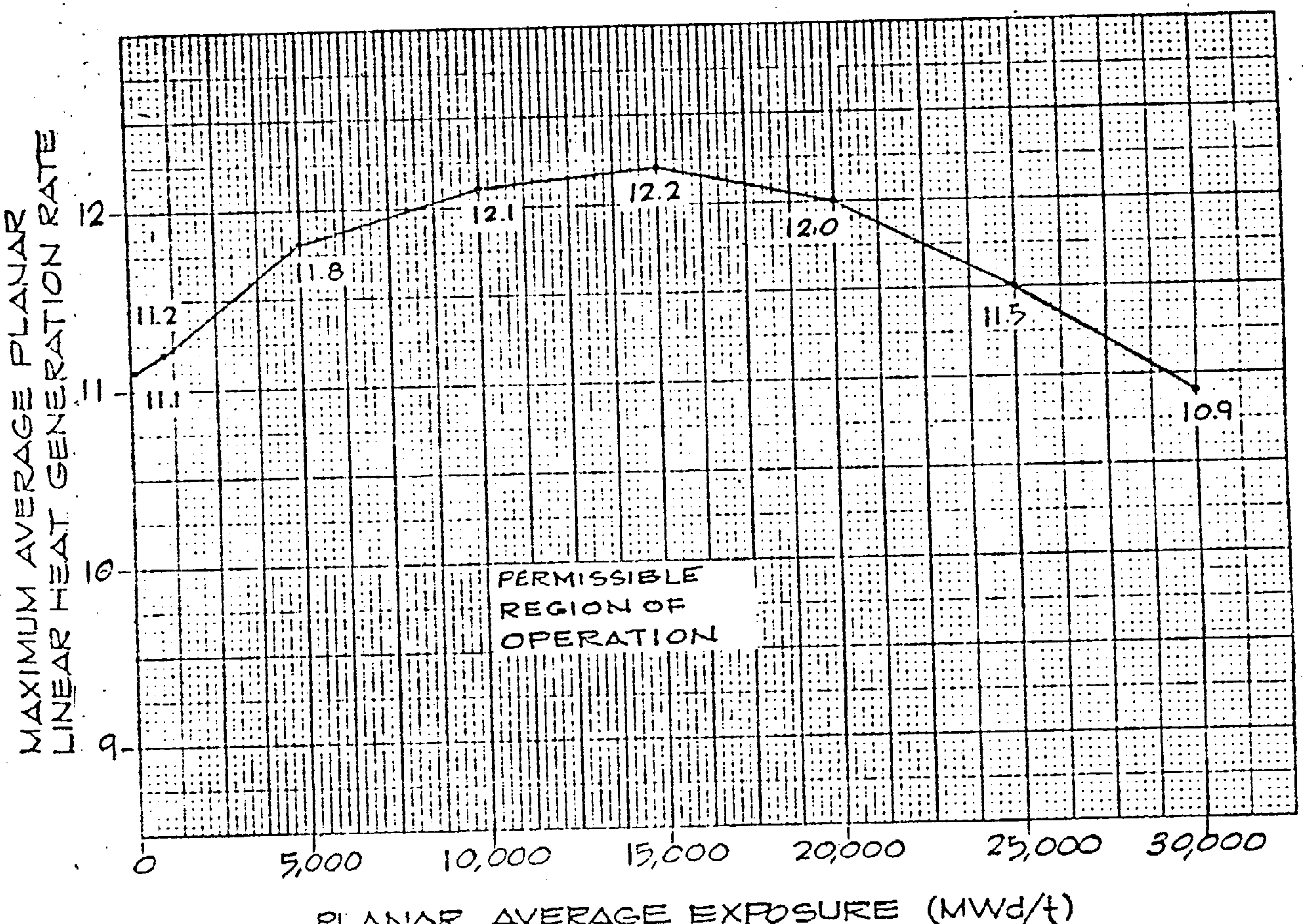



FUEL TYPE 7D230 (7X7)

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE
FIGURE 3.2-1c



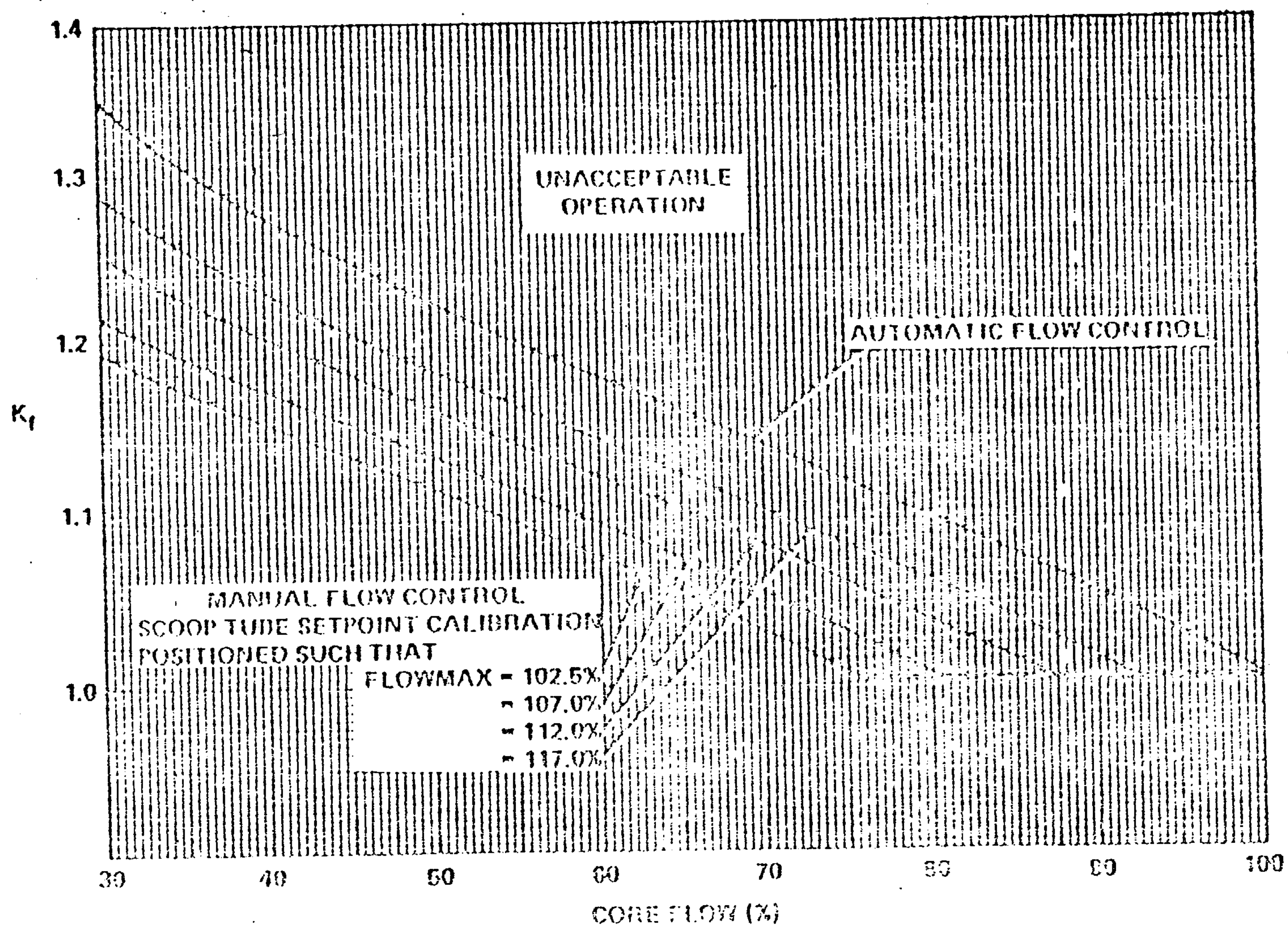

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE
FIGURE 3.2-1d




FUEL TYPE 8D274H (8X8)

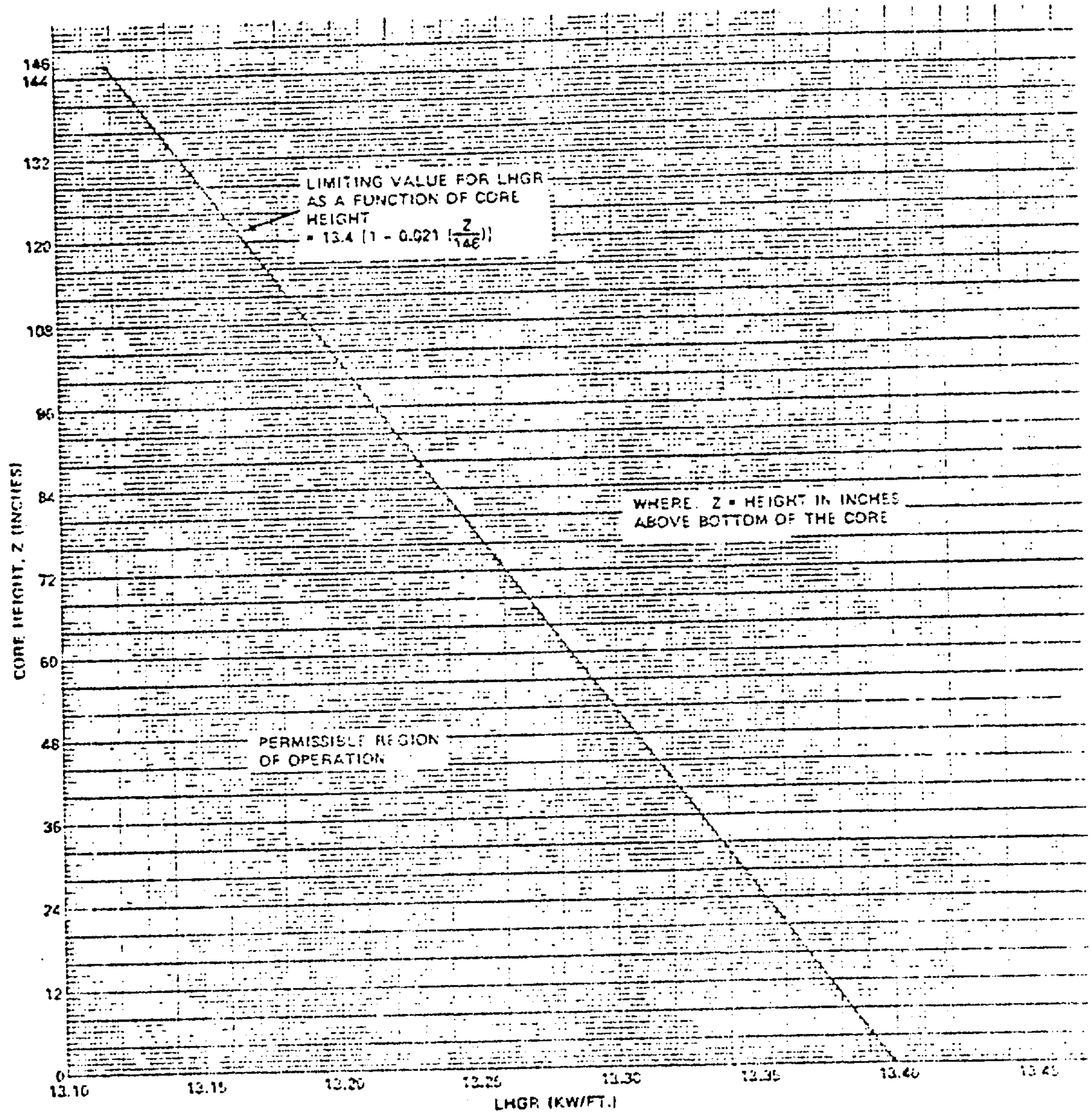
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE

FIGURE 3.2-1e




$K_f$ FACTOR

Figure 3.1-1

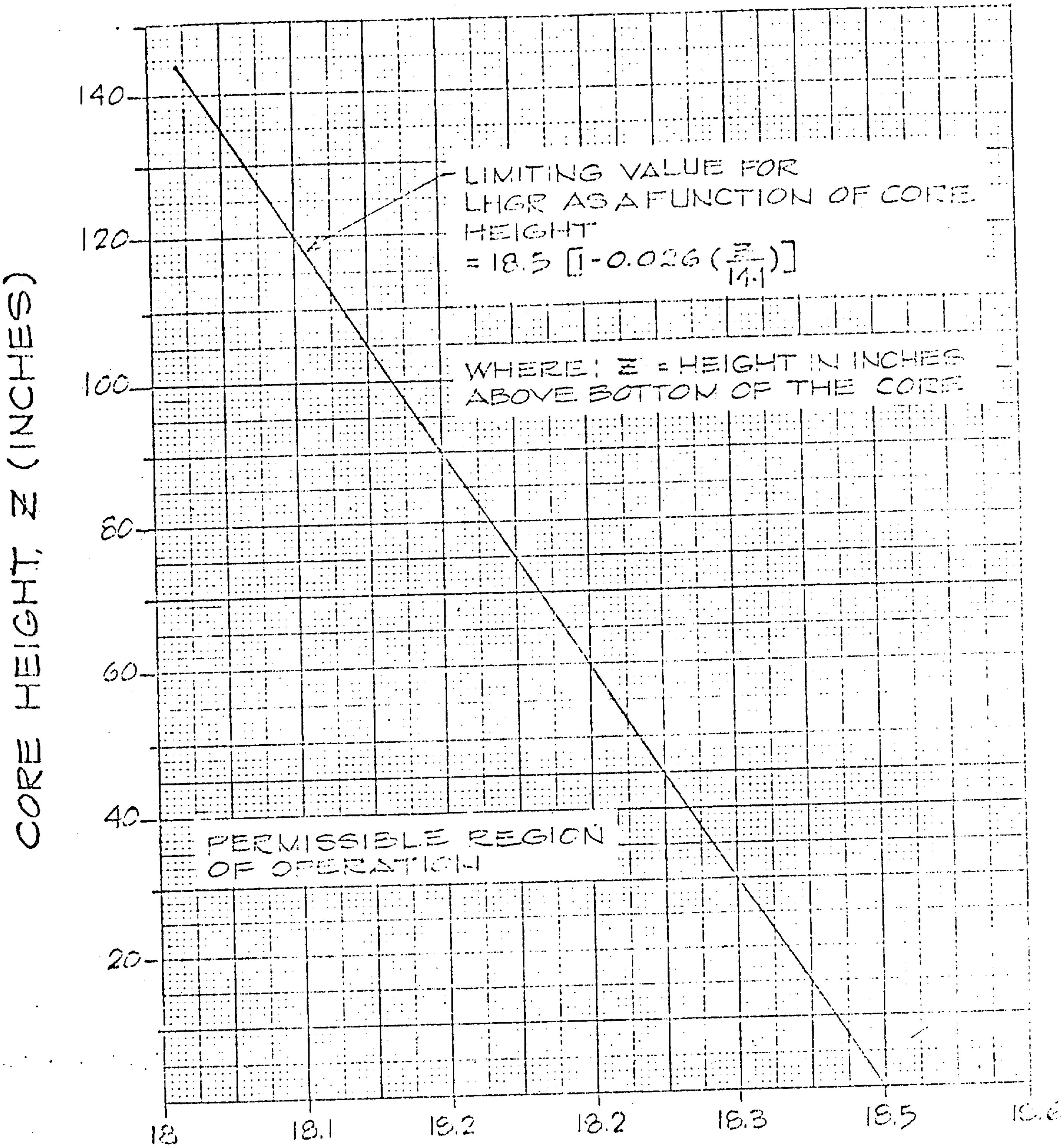



LIMITING VALUE FOR LHGR

8 x 8 Fuel

Figure 3.1-2A

Amendment No.




LHGR (KW/FT)

LIMITING VALUE FOR LHGR

FUEL TYPE 7 X 7

Figure 3.1-2B

Amendment No. 38

Amendment No.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. DPR-62

## CAROLINA POWER AND LIGHT COMPANY

## BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

## DOCKET NO. 50-324

### 1.0 Introduction

By letter dated August 22, 1977, as supplemented September 22, November 10, and 21 , 1977, and by letter dated August 3, 1977, Carolina Power and Light Company (the licensee) requested amendments to Facility Operating License No. DPR-62. By letter dated September 22, 1977, the licensee submitted a reevaluation of the Emergency Core Cooling System performance in compliance with our Order for Modification of License dated March 11, 1977.

The amendments would modify the Technical Specifications for the Brunswick Steam Electric Plant, Unit No. 2 (the facility) to: (1) permit operation of the facility with (a) 8x8 reload fuel bundles, (b) two bypass flow holes drilled in all reload fuel bundles and all initial core fuel remaining in the core after refueling, (c) all initial bypass holes in the core support plate plugged, and (d) limiting maximum average planar linear heat generation rates (MAPLHGR's) as determined by a reevaluation of the Emergency Core Cooling System (ECCS) performance, and (2) permit operation of the facility with modified low pressure permissive setpoints for the RHR and CS pumps, for closing the recirculation pump discharge valves; and for opening the injection valves.

As a result of the licensee's proposal and our review, modification to the licensee's proposed Technical Specifications were necessary. These modifications were discussed with and agreed to by the licensee.

## 2.0 Evaluation

### 2.1 Nuclear Characteristics

The reload information presented in the licensing submittal (Reference 1) closely follows the guidelines of Appendix A of NEDO-20360 (Reference 4). Although NRC staff review of later supplements to this report is not complete, this topical report has been found tentatively acceptable for use in connection with BWR-4 reactors containing 8x8 reload fuel.

A total of 140 8x8 fuel bundles with an average U-235 enrichment of 2.74 wt% will be loaded throughout the core; 40 of the reload fuel bundles contain fuel rods having a high gadolinia content (8D274H) and 100 bundles contain rods having a low gadolinia content (8D274L). The core contains a total of 560 bundles. Thus, 25% of the fuel bundles are being replaced for the reload.

The information in Reference 1 shows that the nuclear characteristics of the cycle 2 core, consisting of both the reload 8x8 fuel and the once burned 7x7 fuel, are very similar to the previous core. Typical nuclear characteristics of the reloaded core are given in Table 5-1 of Reference 1. The void coefficient of reactivity at average voids varies from $-1.19 \times 10^{-3}$ to $-1.16 \times 10^{-3}$ $\Delta k/K/\%V$. The Doppler coefficient, at a fuel temperature of $650^{o}C$, varies from $-1.07 \times 10^{-5}$ to $-1.18 \times 10^{-5}$ $\Delta k/K/^{o}F$. Thus based on our review of the information presented in the Brunswick Unit No. 2 licensing submittal and the generic 8x8 reload topical report, it is concluded that fuel temperature and void dependent behavior of the reconstituted core will not differ significantly from that which has been previously reported for cycle 1 of the Brunswick Unit No. 2 reactor.

The cycle 2 minimum shutdown margin is 1.1%$\Delta k$. This meets the Technical Specification requirement that the core be at least 0.28% $\Delta k$ subcritical in the most reactive operating state with the single most reactive control rod fully withdrawn and with all other rods fully inserted.

The information presented in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by at least 0.028 $\Delta K$ at $20^{o}C$, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

## 2.2 Mechanical Design

The two types of Reload 1 fuel assemblies have the same mechanical design and fuel bundle enrichments as the 8D274L and 8D274H fuel assemblies described in the 8x8 generic reload topical report (Reference 4), except for the drilled bypass flow holes in the fuel bundle lower tie plate.

Sufficient plenum volume has been provided above the fuel stack to assure that the increase in internal pressure caused by fission gas release, when combined with the other mechanical design basis loads, does not cause the stress intensity limits (Reference 4) to be exceeded.

The generic reload topical report (Reference 4) which is under review, has been found acceptable as a guide for use in connection with BWR reactors containing 8x8 reload fuel. On the basis of our review of the generic reload topical report and the reload submittal, we conclude that the Reload 1 fuel for the Brunswick Unit No. 2 reactor has an acceptable mechanical design.

## 2.3 Thermal-Hydraulics

The generic 8x8 reload topical report (Reference 4) and the General Electric Thermal Analysis Basis (GETAB) (Reference 6) are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of GETAB, based on the Minimum Critical Power Ratio (MCPR) concept, was used to establish the:

(1) fuel cladding integrity safety limit,

(2) limiting condition of operation such that the safety limit is not exceeded for normal operation and abnormal operational transients, and

(3) limiting conditions of operation such that the initial conditions assumed in the accident analyses are satisfied.

We have reviewed (Reference 7) the GETAB report and have found it acceptable for use in the above applications for 8x8 and 7x7 fuel assemblies.

The Brunswick Unit No. 2 cycle 2 thermal limits based on the GETAB report and the plant specific information provided by the licensee have been reviewed. Our evaluation of these limits is reported herein.

### 2.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit MCPR is 1.06 for both 7x7 and 8x8 fuel types. With this safety limit, based on the GETAB statistical analysis, 99.9% of the fuel rods in the core are not expected to experience transition boiling for abnormal operational transients. The uncertainties in the core operating parameters, plant system operating parameters and the GEXL correlation (Reference 1, Table 4-1) when combined with the design relative bundle power histogram for the core (Reference 4, Figure 4-2), form the basis of the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for Brunswick Unit No. 2 during cycle 2 are the same as those used in Table 4-1 of NEDO-20360 (Reference 4).

The generic core selected for the GETAB statistical analysis is a typical 251 inch diameter vessel/764 fuel assemblies core. The generic GETAB statistical analysis results are conservative since the core bundle power histogram used for the GETAB application is clearly skewed more to the high power side than the actual operating power distributions expected during the second cycle of operation of Brunswick Unit No. 2. This results in a conservative value of the safety limit MCPR which satisfies the 99.9% criterion.

We conclude that the proposed fuel integrity safety limit MCPR of 1.06 is acceptable for both the 7x7 and reload 8x8 fuel in the Brunswick Unit No. 2 reactor core during cycle 2.

### 2.3.2 Operating Limit MCPR

Various transient events will reduce the operating MCPR. To assure that the fuel cladding safety limit MCPR of 1.06 is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which results in the largest reduction in the critical power ratio (i.e., $\Delta$MCPR). The licensee has submitted (References 1, 2) the results of analyses of those transients which produce a significant decrease in MCPR. The types of anticipated abnormal operational transients evaluated were load rejection without bypass, feedwater temperature decrease, rod withdrawal error, etc.

The most limiting abnormal operational transient from rated conditions in these categories for the 8x8 fuel was the load rejection with failure of the bypass valves, and for the 7x7 fuel was the rod withdrawal error.

The maximum ΔMCPR's for the 7x7 fuel and the 8x8 fuel which resulted from this transient analysis (assuming at least 104% of rated core power , end of cycle 2 burnup, and 100% of rated core flow) were 0.20 and 0.24 respectively.

Addition of these ΔMCPR's to the safety limit MCPR (1.06) gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur. Therefore, the maximum operating limit MCPR's are 1.26 for 7x7 fuel and 1.30 for 8x8 fuel, at rated core flow conditions.

The transient analyses include Design Conservatism Factors (DCF) of 0.80, 1.25, and 0.95 for the scram reactivity functions, void coefficient, and Doppler coefficient respectively. Until the generic review on the DCF's is complete, use of the above values in conjunction with other conservatisms, are considered acceptable. The initial MCPR's and initial conditions assumed in the transient analyses were equal to, or conservatively greater than, the established operating values. Thus, the combination of the above DCF's and other conservatisms used in the analyses provide conservative margins which are acceptable to the NRC staff.

The above operating limit MCPR's,at rated flow, will assure that the fuel cladding integrity safety limit will not be exceeded during any anticipated abnormal operational transient during cycle 2 operations. Thus the above stated operating MCPR's are acceptable for the Brunswick Unit No. 2 reactor during cycle 2 operations.

2.3.3 Generator Load Rejection Without Bypass

The anticipated operational transient which causes the most severe reactor isolation is the generator load rejection without bypass. Fast closure of the turbine control valves therefore produces a large pressure increase in the reactor.

By letter dated September 20, 1977 (Reference 16), the licensee provided new analysis based on revised safety-relief valve setpoints which supersedes the analysis provided in reference 1. Amendment No. 31 dated October 6, 1977 (Reference 17) approves the new safety relief valve setpoints. For discussion of safety considerations with the new safety-relief valve setpoints see the safety evaluation accompanying reference 17.

### 2.3.4 Rod Withdrawal Error

The rod withdrawal error transient (RWE) is discussed in References 1 and 2 for worst case conditions. The rod withdrawal error analysis is based on the most reactive reactor state and conservatively assumes no xenon, which maximizes the amount of excess reactivity inserted upon withdrawal of the maximum worth control rod from the core. The analysis also allows for the most severe rod block monitor detector failure configuration allowed by the Technical Specifications. The event description and analysis assumptions for the RWE are given in Reference 4. These references indicate that the local power range monitors (LPRM's) will detect and alarm a high local power condition. However, if the reactor operator ignores the LPRM alarm, the rod block monitor (RBM) subsystem (set at 106% of full rated power at 100% core flow) will terminate the RWE transient in time to limit the maximum change in the critical power ratio to 0.20 for 7x7 fuel and 0.165 for 8x8 fuel. A RBM rod block occurring at 106% power and full core flow results in a peak linear heat generation rate of 20.2 kw/ft and 13.4 kw/ft for 7x7 and 8x8 fuels respectively. These calculated LHGR's assure performance below the safe acceptable fuel design limits for 7x7 and 8x8 fuels respectively and are therefore acceptable.

### 2.3.5 Operating MCPR Limits for Less than Rated Flow

To assure that the safety limit MCPR is not violated for the limiting flow increase transient (recirculation pump speed control failure) starting from less than rated flow conditions, the licensee will operate Brunswick Unit No. 2 in conformance with the limiting conditions for operation as stated in paragraph 3.2.3 of the Technical Specifications. This requires that for core flow rates less than full rated flow, the licensee shall maintain the MCPR above the minimum operating values.

The minimum MCPR values for less than full rated flow are equal to the MCPR for full rated flow multiplied by the respective $K_f$ factor values appearing in Figure 3.1-1 of the Technical Specifications.

The $K_f$ factor curves were generically derived and assure that for the most limiting flow increase transients, occurring from less than rated core flow, the actual MCPR will not exceed the safety limit MCPR of 1.06.

Application of the above stated $K_f$ factors for reduced flow conditions results in calculated consequences for the limiting anticipated flow increase transients which do not exceed the thermal limits of the fuel or the pressure limits of the reactor coolant boundary.

## 2.4 Accident Analysis

Our evaluation of postulated accidents affected by the actions being considered are discussed in the following sections.

### 2.4.1 ECCS Appendix K Analysis

In December of 1976 the NRC staff was informed that certain input errors and computer code errors had been made in the evaluations of ECCS performance for Brunswick Unit No. 2. An Order was issued to Carolina Power and Light Company on March 11, 1977 (Reference 10), requiring that corrected "revised calculations fully conforming to the requirements of 10 CFR 50.46 are to be provided for the Brunswick Unit No. 2 facility as soon as possible." Such corrected analyses were provided for the present reload in Reference 3. The corrected analyses included correction of all input errors previously made and correction of all computer code errors. The corrected analyses were performed using a calculational model which contains several model changes approved by the NRC staff in a Safety Evaluation issued April 12, 1977 (Reference 14).

We have reviewed the corrected analyses submitted for Reload 1 in Reference 3. We conclude that the Brunswick Unit No. 2 will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: 1) it is operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Figures 3.2-1a, 1b, 1c, 1d, and 1e of Reference 3; and 2) when it is operated at a Minimum Critical Power Ratio (MCPR) equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the LOCA, as described in Section 2.3.2 of this Safety Evaluation). The analyses submitted in Reference 3 and

Reference 13 references James A. FitzPatrick Nuclear Power Plant as the lead plant, (Reference 15) (i.e., complete break spectrum study) submitted with the corrected model. Brunswick Unit No. 2 is a BWR/4 with low pressure coolant injection (LPCI) System modification which is the same class of plant as James A. FitzPatrick Nuclear Power Plant. The analyses provide all information requested for non-lead plants in the NRC letter to GE on June 30, 1977 (Reference 12), regarding number of breaks to be analyzed, documentation to be provided, etc., for the new analyses.

The analysis showed that the particular break producing the highest peak clad temperature (PCT) is a recirculation pump discharge line break having an area approximately 80% as large as the largest discharge line break. That break for Brunswick Unit No. 2 is herein called the limiting break. This is the same limiting break as was found for the lead plant. Reasons why this break's analysis for this class of plant produces the highest PCT are presented below.

The limiting location is the recirculation pump discharge line rather than the larger diameter recirculation pump suction line due to the LPCI system modification previously made on this class of plants. The LPCI modification consisted of eliminating the loop-selection-logic system which previously had been provided to select the unbroken recirculation line following a LOCA and direct all LPCI flow from both LPCI systems to the unbroken recirculation line. (The loop-selection-logic system was subject to single failures, such as failure to open of the single LPCI discharge valve leading to the unbroken recirculation line. This failure would prevent all LPCI flow from both LPCI systems from entering the reactor). In place of the loop-selection-logic system, one LPCI system was permanently piped to one recirculation pump discharge line, and the other LPCI system was permanently piped to the second recirculation pump discharge line. After blowdown following a LOCA, the recirculation pump discharge valves close. These valves are located between the LPCI system injection point on the recirculation pump discharge line and any potential break location on the recirculation pump suction line. The LPCI system connected to the broken recirculation line is thus isolated from any suction line break (the other LPCI system is also isolated because of its connection to the unbroken line), and since only one LPCI loop can be disabled by any single failure, the largest (suction line) break can derive credit for earlier reflooding due to effectiveness of at least one LPCI system. This significantly reduces PCT calculated for the large (suction line) break, and, for plants with the LPCI modifications, reduces it below PCT calculated for the smaller discharge line break. For the discharge line break, the LPCI system injection point cannot be isolated from the break location. Therefore, a break in a smaller diameter line than the suction line for plants without the LPCI modification would be expected to yield a lower PCT. For plants with the LPCI modifications, as with Brunswick Unit No. 2, lack of LPCI flow* for the discharge line break delays the reflooding (with respect to the suction line break where LPCI flow from at least one system is available). This condition results in the discharge break for Brunswick Unit No. 2 being limiting. This result (discharge break limiting) has been observed previously and in fact was the reason behind design and implementation of the LPCI modification. (A MAPLHGR limit increase is realized by lowering of the previously limiting suction line break PCT). The analysis for the lead plant, James A. FitzPatrick (reference 15), also showed the 0.8 discharge line break to be limiting.

For Brunswick Unit No. 2, calculations for the largest suction line break, and the largest discharge break have also been provided. These calculations showed PCT's lower than the limiting break PCT.

The determination that for various size discharge line breaks the 0.8 discharge line break is limiting was supported by studies showing that for Brunswick Unit No. 2 this size break results in the longest period during which the hot node remains uncovered. The lead plant analyses (reference 15) confirmed this procedure for determining break size. The analysis as to the limiting break size in Reference 15 is incorporated in this evaluation by reference.

The licensee by letter dated August 3, 1977 requested revision of the Technical Specifications for Brunswick Unit No. 2 to make the following revisions to the low pressure permissive set points:

(1) A change from 325 to 410 psig in the low pressure permissive set point for starting the RHR and CS pumps and for opening the injection valves; and

(2) A change from 325 to 310 psig in the low pressure permissive setpoint for closing the recirculation pump discharge valves.

The first change (#1 above) is acceptable since it results in earlier availability of ECCS equipment following a postulated LOCA; therefore the existing ECCS analyses, which assume the presently existing (later) availability of that equipment, will be slightly more conservative once the change is made.

*One LPCI system cannot be isolated from the break and its flow is lost out the break; a single failure is assumed in the other system.

The second change (#2 above) is also acceptable for implementation because the LOCA analysis for cycle 2 assumes a 285 psig to 335 psig pressure permissive set point, and the proposed set point (310 psig) is within this range.

Based on the foregoing, we conclude that this ECCS reevaluation fully meets the requirements of 10 CFR 50.46 and thereby satisfies the conditions of our Order for Modification of License dated March 11, 1977.

### 2.4.3 Fuel Loading Error

The fuel loading error is discussed in Reference 1 for 8x8 fuel bundles placed in an improper location or rotated 180 degrees in a location near the center of the core. The information in Reference 1 indicates that a fuel loading error results in a peak linear heat generation rate (LHGR) of 16.7 kw/ft in the misloaded 8x8 fuel bundle. The calculated peak LHGR is below that required to exceed the 1% plastic strain fuel design limit. The resulting MCPR is 1.07 which compares with a safety limit MCPR of 1.06. Fuel bundles adjacent to the misloaded bundle are insignificantly affected.

We find the analyses and results of the fuel loading error acceptable.

### 2.4.4 Control Rod Drop Accident

The cycle 2 control rod drop accident for Brunswick Unit No. 2 is within the generic bounding analysis presented in Reference 4. The actual cycle 2 Doppler coefficient for the cold and hot startup conditions conservatively falls within the values assumed in the bounding analysis. The accident reactivity shape functions for both hot and cold startup conditions falls within the bounding analysis. The resultant peak enthalpies from the generic bounding analysis for the cold and hot startup cases were calculated to be less than the 280 cal/gm design limit.

The scram reactivity functions are outside the bounding analyses for high reactivity values, but are bounded up to a total negative scram insertion of 0.02 $\Delta k$. The combined Doppler and 0.02 $\Delta k$ scram will be more than sufficient to terminate the accident and bring the reactor core subcritical for control rod worths of interest. The peak fuel enthalpy will not exceed 280 cal/gm. We find this result acceptable.

## 2.5 Overpressure Analysis

The licensee presented the results of an overpressure analysis to demonstrate that an adequate margin exists to the ASME code allowable vessel pressure, which is 110% of the vessel design pressure. The transient analyzed was the fast closure of all main steamline isolation valves with the conservative assumption that a reactor scram would occur on the second (high neutron flux) scram signal rather than the first (10% valve closure position switches).

By letter dated September 20, 1977 (reference 16), the licensee provided new analyses based on revised safety-relief valve setpoints which supersedes the analysis provided in reference 1. Amendment No. 31 to Operating License No. DPR-62 dated October 6, 1977 (reference 17) approves the new safety relief valve setpoints. For discussion of the safety considerations of the overpressure analyses with the new set-points see the safety evaluation accompanying reference 17.

## 2.6 Thermal Hydraulic Stability Analysis

The thermal hydraulic stability analyses and results are described in Reference 1. The results of the cycle 2 analysis show that the 7x7 and 8x8 channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is within the operational design guide in terms of decay ratio. Calculations were also performed by the licensee to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results of this analysis showed that the reactor core stability characteristics at both conditions are within the operational design guide. These results are acceptable to the NRC staff.

We have expressed generic concerns regarding the least stable reactor condition allowed by Technical Specifications. This condition could be reached during an operational transient from high power where the plant sustains a trip of both recirculation pumps. The concerns are motivated by increasing decay ratios in reload fuel cycles and improved fuel design.

Our concerns relate to both the consequences of operating at an ultimate decay ratio and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing our concerns through meetings, topical reports, and a test program.

A reactor core stability test program has been performed at Peach Bottom Unit No. 2 end of Cycle 2.

The test program is expected to be a significant aid in the resolution of our generic concerns on stability. The testing was performed during April 1977. The results from the testing will be provided to the NRC staff by the General Electric Company. The results will be used to refine the reactor stability analysis safety margins.

Until this issue has been resolved generically, the licensee will be required to restrict operations in the natural circulation flow mode to intervals of no more than 12 hours duration, or be shutdown within the next 12 hours. The licensee has agreed to this Technical Specification limitation. The restriction will provide a significant increase in the reactor core stability margins during cycle 2. On the basis of the foregoing, we consider the thermal-hydraulic stability to be acceptable.

### 2.6.1 Recirculation Pump Startup from the Natural Circulation Operational Mode

During recent BWR reload reviews, the question of recirculation pump startup from the natural circulation operational mode was raised. The pump startup could increase flow, collapse moderator voids, and subsequently result in a reactivity insertion transient. The consequences of such an accident sequence has not been previously evaluated, so that for this reload review, additional information was requested.

The licensee provided analyses and startup test results which showed that the startup of recirculation pumps from the natural circulation condition at 40% power and 30% flow resulted in about a 3% power increase (Reference 18). In addition the licensee agreed to a Technical Specification restriction which limits to 12 hour intervals the operating mode under which such pump restart would be possible. We find this measure to be acceptable.

## 3.0 Physics Startup Testing

As part of our review of Reload 1 for Brunswick Unit No. 2, the licensee was requested to provide a description of the cycle 2 physics startup test program. In response to that request, the physics startup test program was provided by the licensee in Reference 2. After discussion, the licensee agreed to perform further additional startup tests addressing measurements of critical eigenvalue, power distribution, TIP reproduceability, and core power asymmetry as described in reference 11. The physics startup tests, along with the tests required to assure compliance with the Technical Specifications, provide an acceptable physics startup test program.

## 4.0 Conclusions

We conclude that the reevaluation of the ECCS performance submitted by the licensee meets the requirements of our Order for Modification of License dated March 11, 1977, and based on our evaluation of the applications and the available information and the requirements set forth above, it is acceptable for the licensee to proceed with cycle 2 operation in the manner proposed.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

We have 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 23, 1977

References


1. "General Electric Boiling Water Reactor Reload-1 Licensing Amendment for Brunswick Steam Electric Plant Unit 2", NEDO-24029, Class 1, June 1977. Attachment A to CP&L letter dated August 22, 1977.

2. Carolina Power and Light Company letter (B. J. Furr) to U. S. Nuclear Regulatory Commission (A. Schwencer) "Cycle 2 Operation - NRC Request for Information", November 10, 1977.

3. Carolina Power and Light Company letter (E. E. Utley) to U. S. Nuclear Regulatory Commission (Edson G. Case), "Request for License Amendment", September 22, 1977.

4. "General Electric Generic Reload Licensing Application for 8x8 Fuel," Revision 1, Supplement 4, April 1976, NEDO-20360.

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

6. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," General Electric Company, BWR Systems Department, November 1973, NEDO-10958.

7. "Topical Report Evaluation of General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," by Division of Technical Review, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, September 1974.

8. General Electric letter (John A. Hinds) to U. S. Atomic Energy Commission (Walter Butler) "Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports", NEDO-10958 and NEDE-10958, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," July 1974.

9. General Electric letter (Ivan F. Stuart) to U. S. Nuclear Regulatory Commission (Victor Stello, Jr.). "Code Overpressure Protection Analysis - Sensitivity of Peak Vessel Pressures to Valve Operability," December 23, 1975.

10. U. S. Nuclear Regulatory Commission letter (A. Schwencer) to Carolina Power and Light Company (J. A. Jones) "Brunswick Steam Electric Plant, Units Nos. 1 and 2 - Order for Modification of License", March 11, 1977.

11. Letter, B. J. Furr (CP&L), to A. Schwencer, "Request for Additional Information - Cycle 2", dated November 21, 1977.

12. U. S. Nuclear Regulatory Commission letter (Darrell G. Eisenhut) to General Electric Company (E. D. Fuller) "Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-Lead Plants, June 30, 1977.

13. Power Authority of the State of New York letter (George T. Berry) to Nuclear Regulatory Commission (R. W. Reid) "Additional Responses to Reload Questions, Docket No. 50-333," dated August 25, 1977.

14. NRC letter (K. Goller) to General Electric (G. Sherwood), LOCA Model Changes, April 12, 1977.

15. Letter, George T. Berry (PASNY) to Robert W. Reid (NRC), "James A. Fitzpatrick Nuclear Power Plant ECCS Analysis Docket No. 50-333," dated July 29, 1977.

16. Letter, E. E. Utley (CP&L) to A. Schwencer (NRC), "Request for License Amendment - Revision of Technical Specifications", dated September 20, 1977.

17. Amendment No. 31 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant Unit No. 2, dated October 6, 1977.

18. Letter, B. J. Furr (CP&L), to A. Schwencer (NRC) dated November 21, 1977, re: natural circulation testing.

# UNITED STATES NUCLEAR REGULATORY COMMISSION

## DOCKET NO. 50-324

## CAROLINA POWER AND LIGHT COMPANY

## NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 38 to Facility Operating License No. DPR-62, issued to Carolina Power and Light Company (the licensee), which revised Technical Specifications for operation of Brunswick Steam Electric Plant, Unit No. 2 (the facility) located in Brunswick County, North Carolina. The amendment is effective as of the date of issuance.

The amendment changes the Technical Specifications for the facility to establish revised safety and operating limits for operation in Cycle 2 with both 7x7 and new 8x8 fuel, and includes changes resulting from a reevaluation of Emergency Core Cooling System (ECCS) cooling performance submitted by CP&L on September 22, 1977, in compliance with the Commission's Order for Modification of License dated March 11, 1977. This reevaluation corrected the errors identified in the March 11, 1977 Order and included the effect of other recently approved model changes in the ECCS evaluation models. The CP&L submittal of September 22, 1977, satisfies the action required by the March 11, 1977 Order. Therefore, effective upon issuance of this amendment, the Commission's Order for Modification of License dated March 11, 1977, relative to Facility Operating License No. DPR-62, is terminated.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the Federal Register on September 26, 1977 (42 F.R. 48951) and on September 29, 1977 (42 F.R. 51676). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the applications for amendment dated August 3, August 22, and September 22, 1977, as supplemented on November 10 and 21, 1977, (2) Amendment No. 38 to License No. DPR-62, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Southport-Brunswick County Library, 109 West Moore Street, Southport, North Carolina 28461. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 23rd day of November 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Alfred Burger

Alfred Burger, Acting Chief
Operating Reactors Branch #1
Division of Operating Reactors