Docket No. 50-324

July 12, 1982

Mr. J. A. Jones
Senior Executive Vice President
Carolina Power & Light Company
P.O. Box 1551
Raleigh, North Carolina 27602

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 71 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your submittals of May 20, June 24, and June 28, 1982, and others as referenced in the enclosed Safety Evaluation.

The amendment changes the Technical Specifications to establish revised safety and operating limits for operation of Brunswick Unit No. 2 during fuel Cycle Number 5.

Our evaluation of your core spray sparger crack analysis and corrective actions is included in the Safety Evaluation.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 71 to DPR-62
2. Safety Evaluation
3. Notice

cc: w/enclosures
See next page

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DATE	7/9/82	7/9/82	7/9/82	7/9/82	7/12/82		

Mr. J. A. Jones
Carolina Power & Light Company

cc:

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. DPR-62

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

   A. The application for amendment by Carolina Power & Light Company dated May 20, 1982 and supplements dated June 24, 1982 and June 28, 1982 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-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. 71 , 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

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 12, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 71

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Remove the following pages and replace with identically numbered pages.

I
II
1-1
1-2
1-3
1-4
1-5
1-6
Add 1-6a
3/4 1-1
3/4 2-1
3/4 2-2
3/4 2-3
3/4 2-4
3/4 2-5
3/4 2-6
3/4 2-7
3/4 2-8
3/4 2-9
Add 3/4 2-9a
3/4 2-10
3/4 2-11
Add 3/4 2-11A
3/4 2-12
Add 3/4 2-12A
Add 3/4 2-12B
3/4 3-42
3/4 10-3
B3/4 2-3

SECTION

1.0 DEFINITIONS

SECTION

1.0 DEFINITIONS (Continued)

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

ACTIONS are those additional requirements specified as corollary statements to each specification and shall be part of the specifications.

AVERAGE PLANAR EXPOSURE

The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment as necessary of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indication and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABLITY including alarm and/or trip functions.

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

CORE ALTERATION

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

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be concentration of I-131, μCi/gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The following is defined equivalent to 1μCi of I-131: I-132, 29μCi; I-133, 3.6μCi; I-134, insignificant; I-135, 12μCi.

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

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

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

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

a. Turbine stop valves, and

b. Turbine control valves.

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

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

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

ISOLATION SYSTEM RESPONSE TIME

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable.

LIMITING CONTROL ROD PATTERN

A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

LINEAR HEAT GENERATION RATE (LHGR) shall be the power generation in an arbitrary length of fuel rod, usually one foot. It is the integral of the heat flux over the heat transfer area associated with the unit length, usually measured in KW/ft.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all relays and contacts of a logic circuit, from sensor output to activated device, to ensure that components are OPERABLE.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

MAXIMUM FRACTION OF LIMITING POWER DENSITY shall be the highest value of LINEAR HEAT GENERATION RATE (LHGR) divided by the corresponding LHGR limit occurring in the reactor core.

MAXIMUM TOTAL PEAKING FACTOR

The MAXIMUM TOTAL PEAKING FACTOR (MTPF) shall be the largest TPF which exists in the core for a given class of fuel for a given operating condition.

MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

ODYN OPTION A

ODYN OPTION A shall be analyses which refer to minimum critical power ratio limits which are determined using a transient analysis plus an analysis uncertainty penalty.

ODYN OPTION B

ODYN OPTION B shall be analyses which refer to minimum critical power ratio limits determined using a transient analysis which includes a requirement for 20% scram insertion times to reduce the analysis uncertainty penalty.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION

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

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and are 1) described in Section 13 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

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

PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

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

b. All equipment hatches are closed and sealed.

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

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

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

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2436 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REPORTABLE OCCURRENCE

A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of notches. All rods fully inserted are equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All automatic Reactor Building ventilation system isolation valves or dampers are OPERABLE or secured in the isolated position.
b. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.1.
c. At least one door in each access to the Reactor Building is closed.
d. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor would be subcritical assuming that all control rods capable of insertion are fully inserted except for the analytically determined highest worth rod which is assumed to be fully withdrawn, and the reactor is in the shutdown condition, cold, 68°F, and Xenon-free.

SPIRAL RELOAD

A SPIRAL RELOAD is the reverse of a SPIRAL UNLOAD. Except for two diagonal fuel bundles around each of the four SRMs, the fuel in the interior of the core, symmetric to the SRMs, is loaded first.

SPIRAL UNLOAD

A SPIRAL UNLOAD is a core unload performed by first removing the fuel from the outermost control cells (four bundles surrounding a control blade). Unloading continues in a spiral fashion by removing fuel from the outermost periphery to the interior of the core, symmetric about the SRMs, except for two diagonal fuel bundles around each of the four SRMs.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.
b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TOTAL PEAKING FACTOR

The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4 1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than 0.38% Δ k/k.

APPLICABILITY: CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than 0.38% Δ k/k:

a. In CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In CONDITION 3 or 4, immediately verify all control rods to be fully inserted, suspend all activities that could reduce the SHUTDOWN MARGIN, and demonstrate SECONDARY CONTAINMENT INTEGRITY within 1 hour; reestablish the required SHUTDOWN MARGIN.

c. In CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN, fully insert all insertable control rods, and demonstrate SECONDARY CONTAINMENT INTEGRITY within 1 hour; reestablish the required SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.1 THE SHUTDOWN MARGIN shall be determined to be equal to or greater than 0.38% Δ k/k:

a. By measurement within 24 hours prior to or during the first start-up after completing CORE ALTERATIONS, and

b. By analytical determination within 12 hours after detection of a withdrawn control rod that is immovable or untrippable, except that the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable rod.

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 following limits:

a. During two recirculation loop operation, the limits are shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7, 3.2.1-8, or 3.2.1-9.*

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% of RATED THERMAL POWER.

ACTION: With an APLHGR exceeding the limits of Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7, 3.2.1-8, or 3.2.1-9,*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

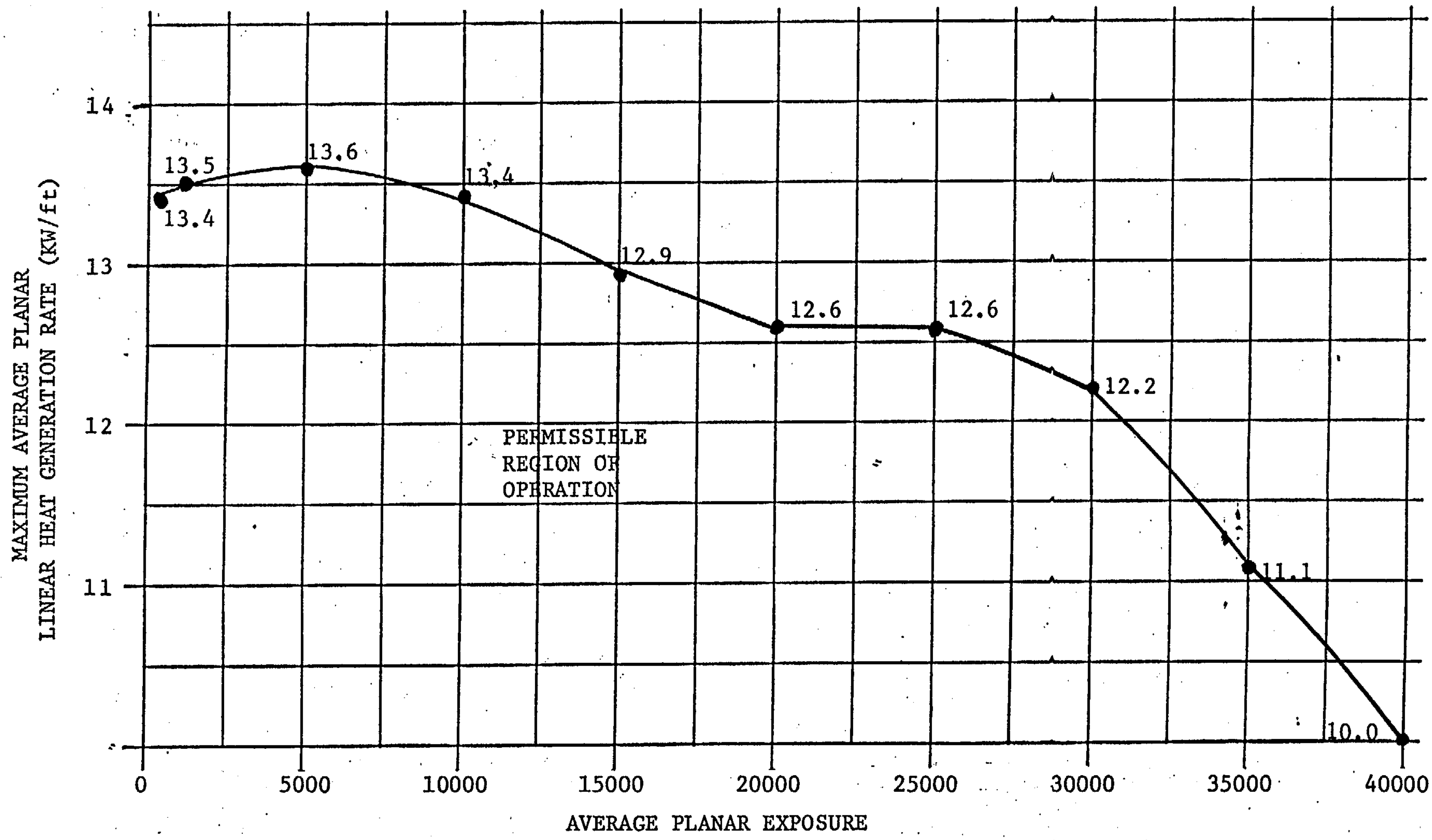
4.2.1 All APLHGR's shall be verified to be equal to or less than the applicable limit determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7, 3.2.1-8 or 3.2.1-9:*

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.

* These limits differ from those found in General Electric reports Y1003J01A37, March 1982, "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant, Unit 2, Reload 4," (with and without RPT) due to a MAPLHGR reduction penalty imposed by NRC due to core spray sparger cracking.

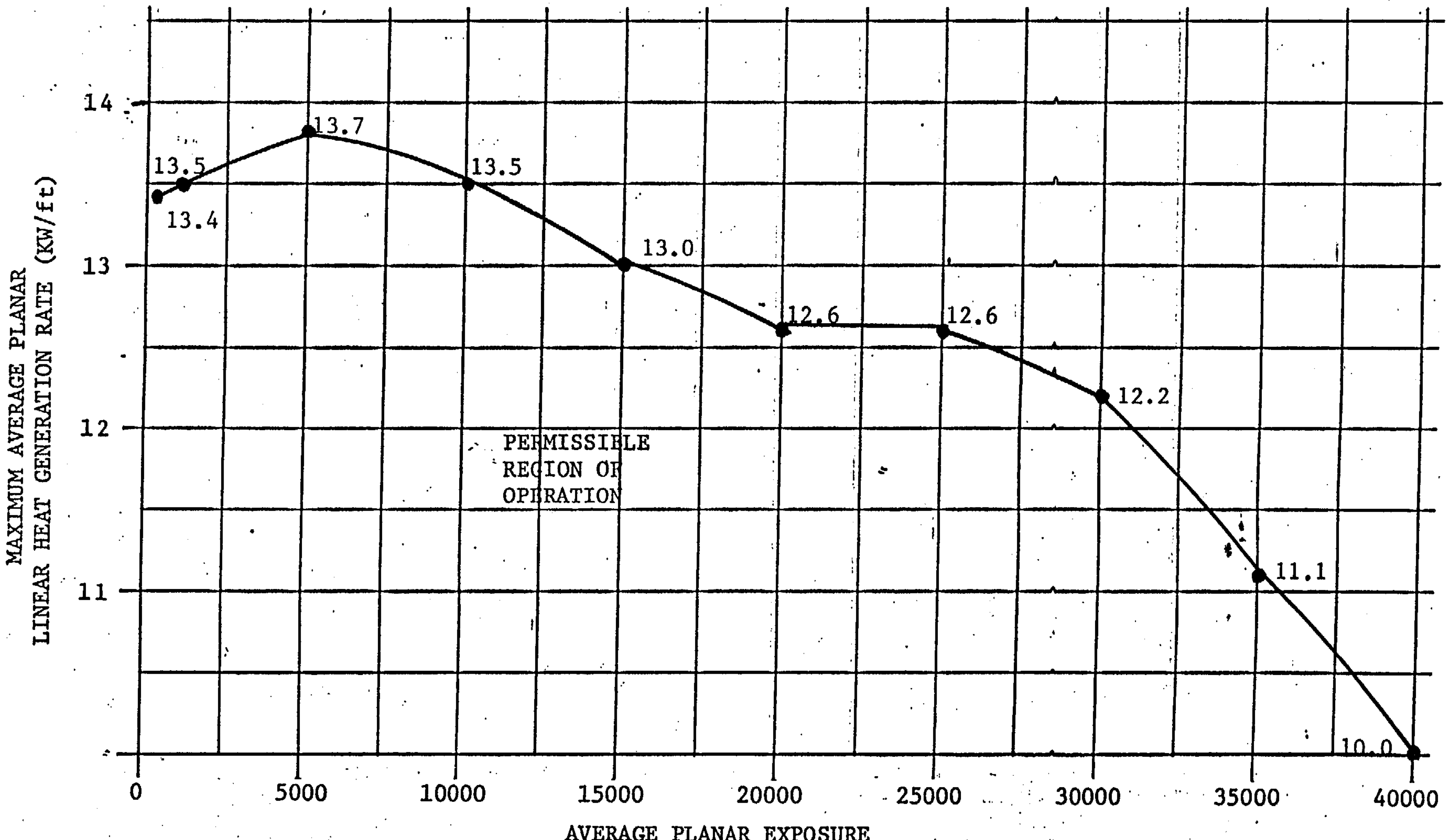


FUEL TYPE 1 & 2 (7x7)

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

FIGURE 3.2.1-1

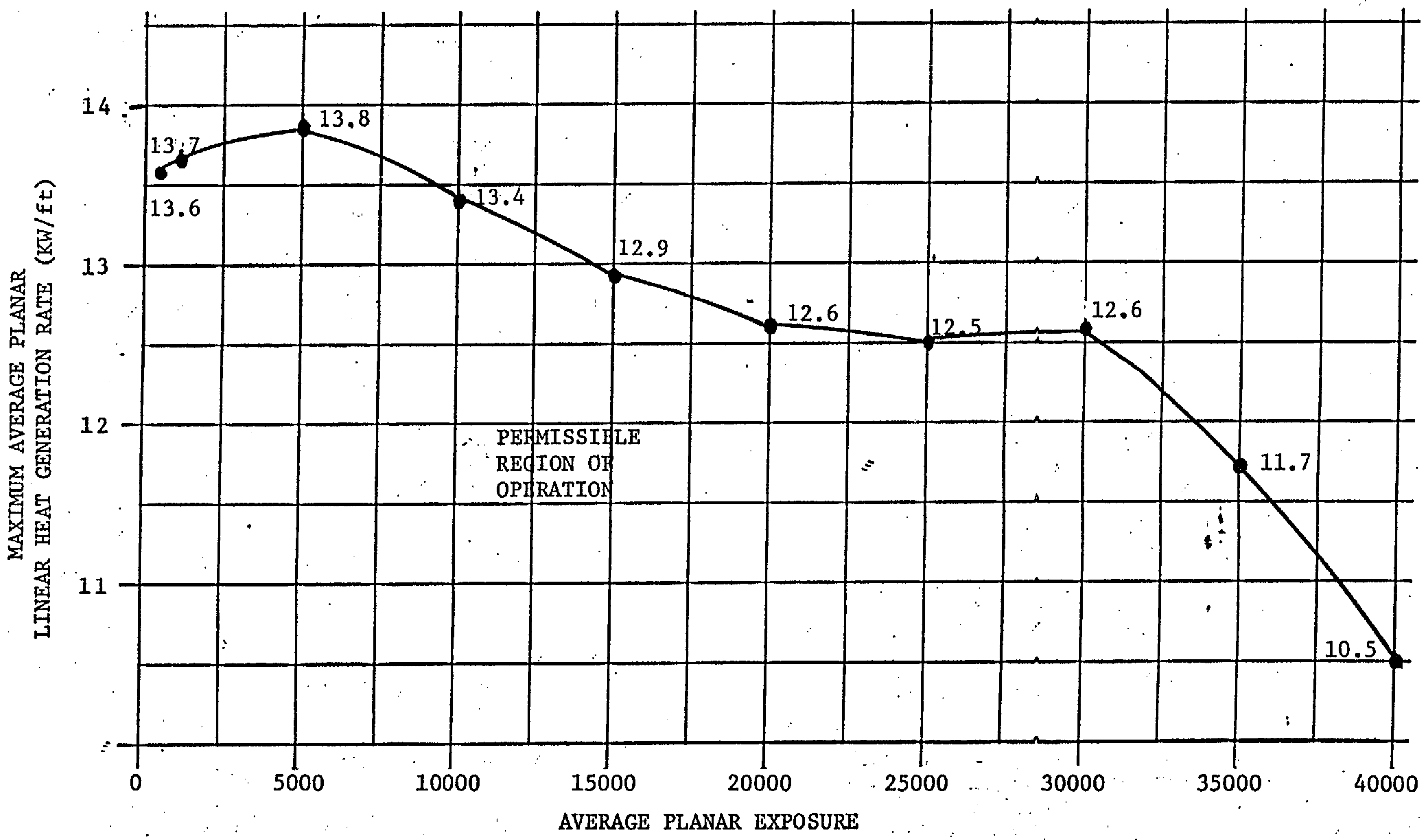
NOTE: See Footnote for Specification 3.2.1



FUEL TYPE 3 (7x7)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-2

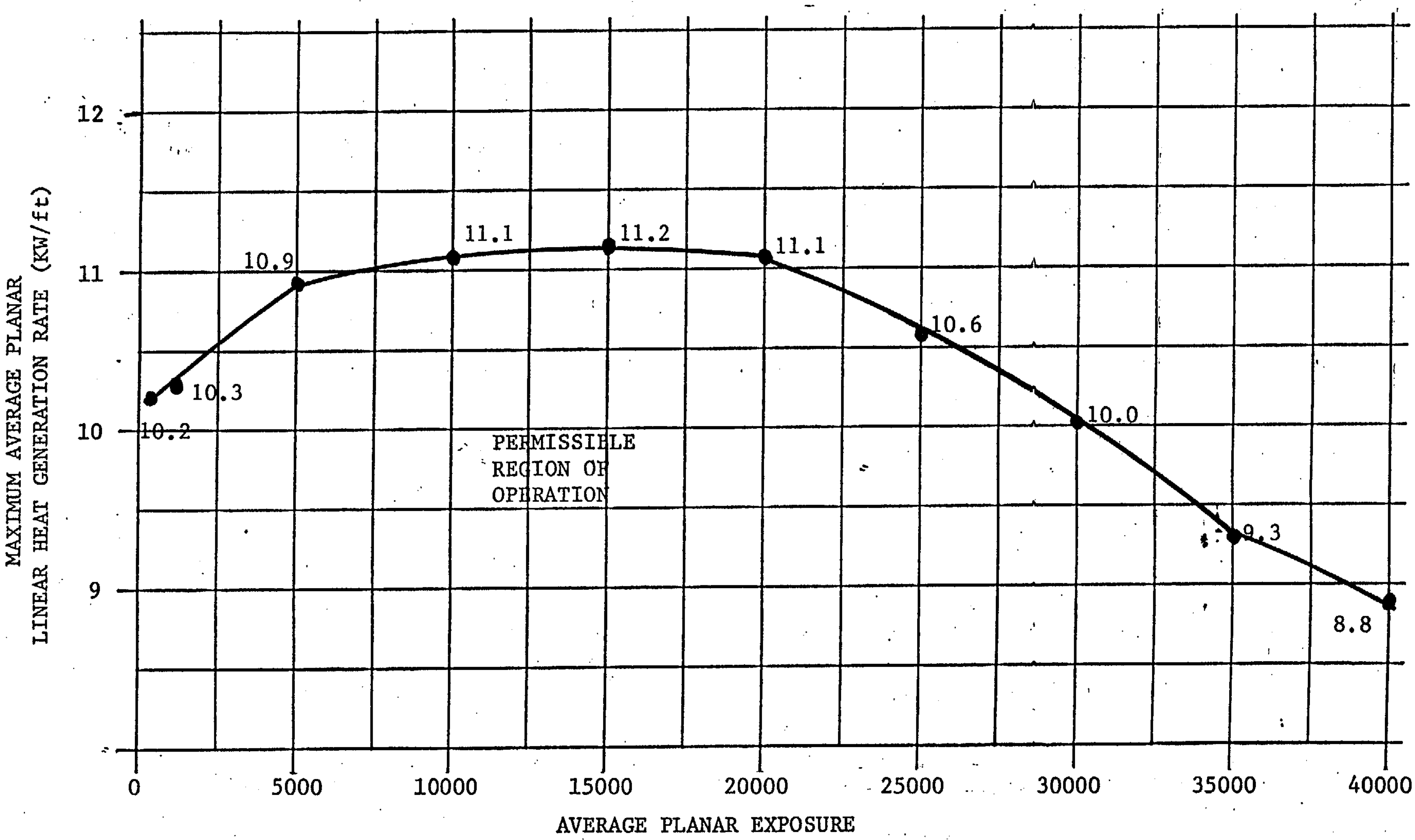
NOTE: See Footnote for Specification 3.2.1



FUEL TYPE 7D230 (7x7)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-3

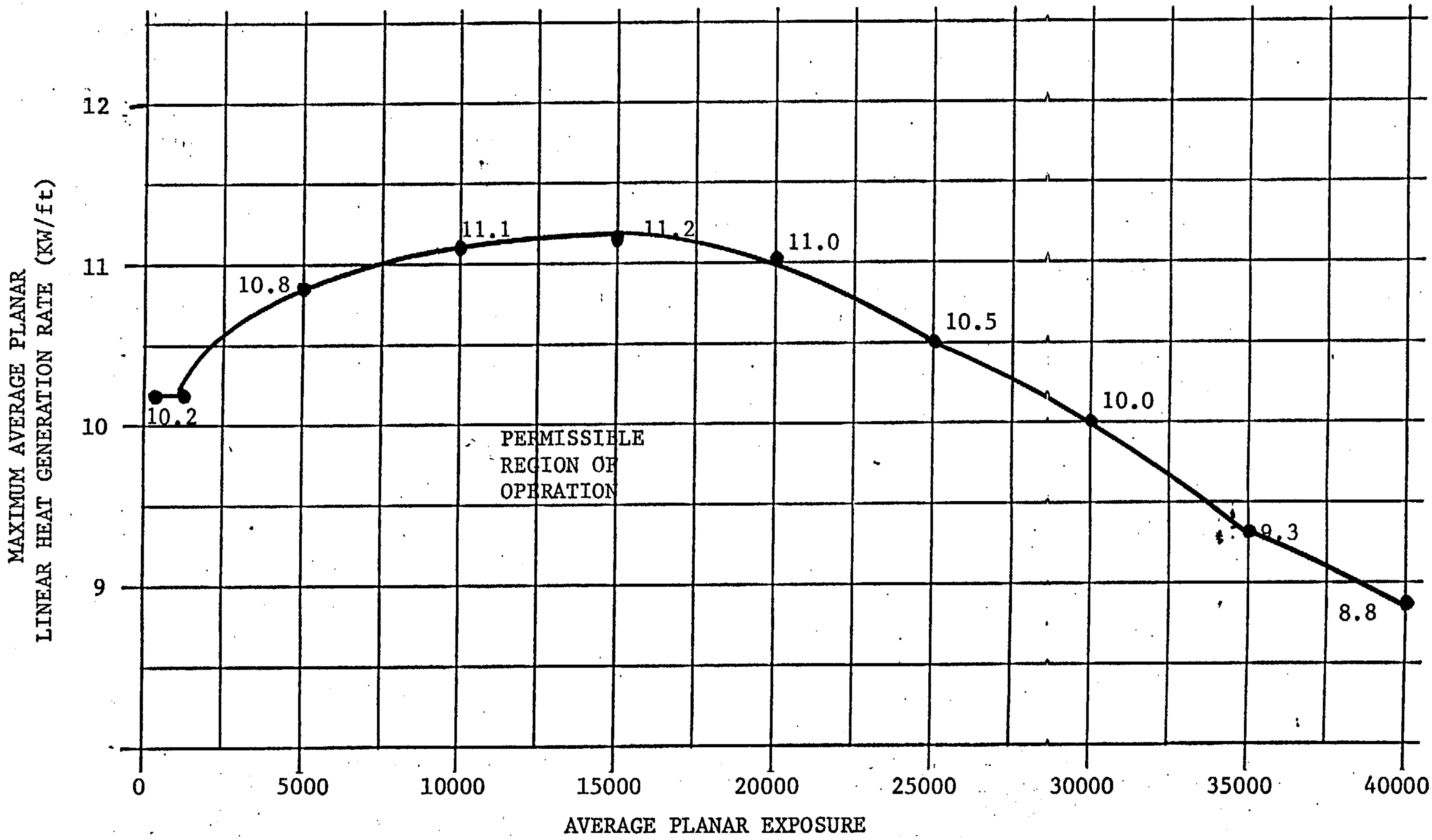
* NOTE: See Footnote for Specification 3.2.1.



FUEL TYPE 8D274L (8x8)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-4

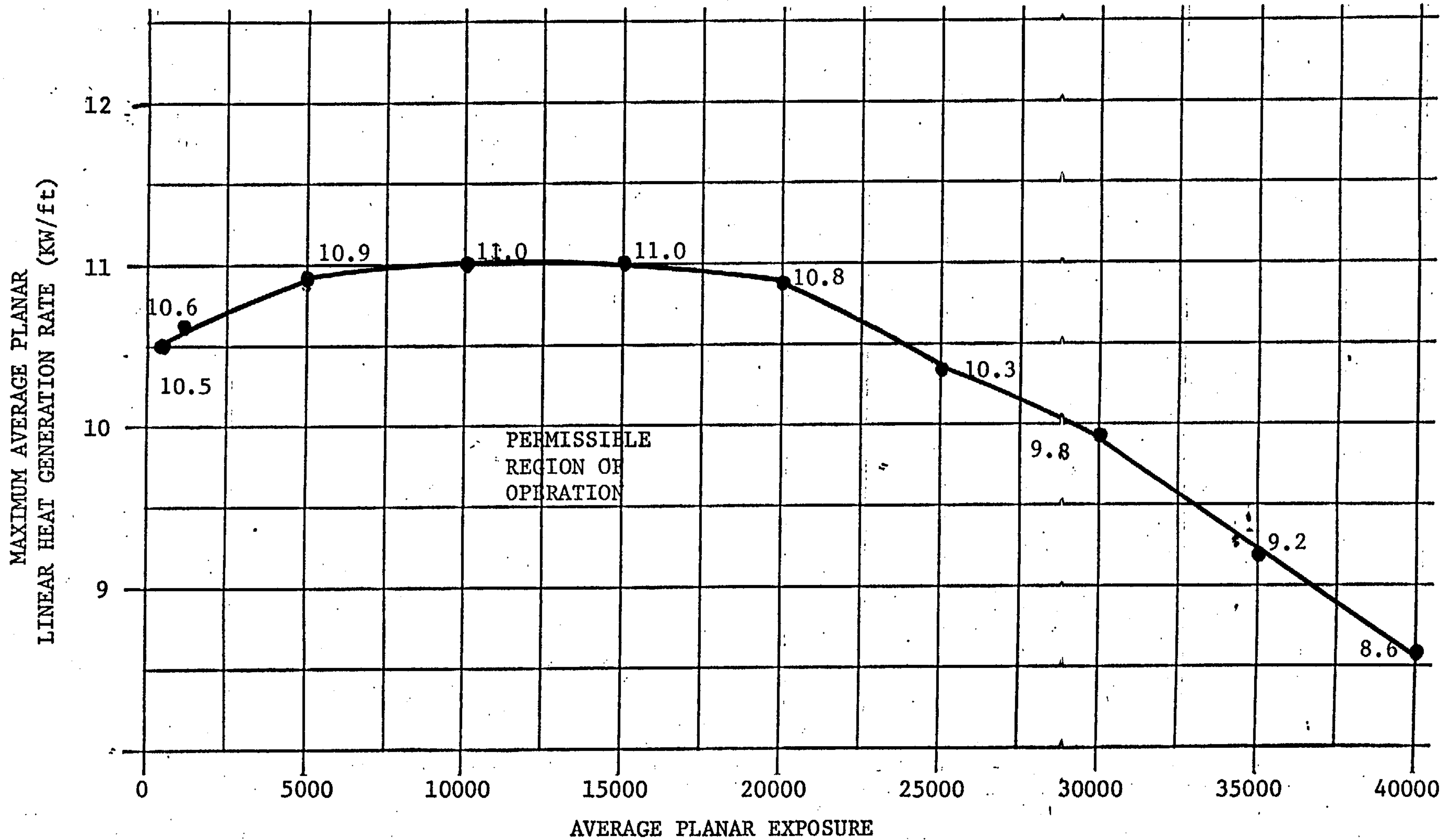
NOTE: See Footnote for Specification 3.2.1.



FUEL TYPE 8D274H (8x8)
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-5

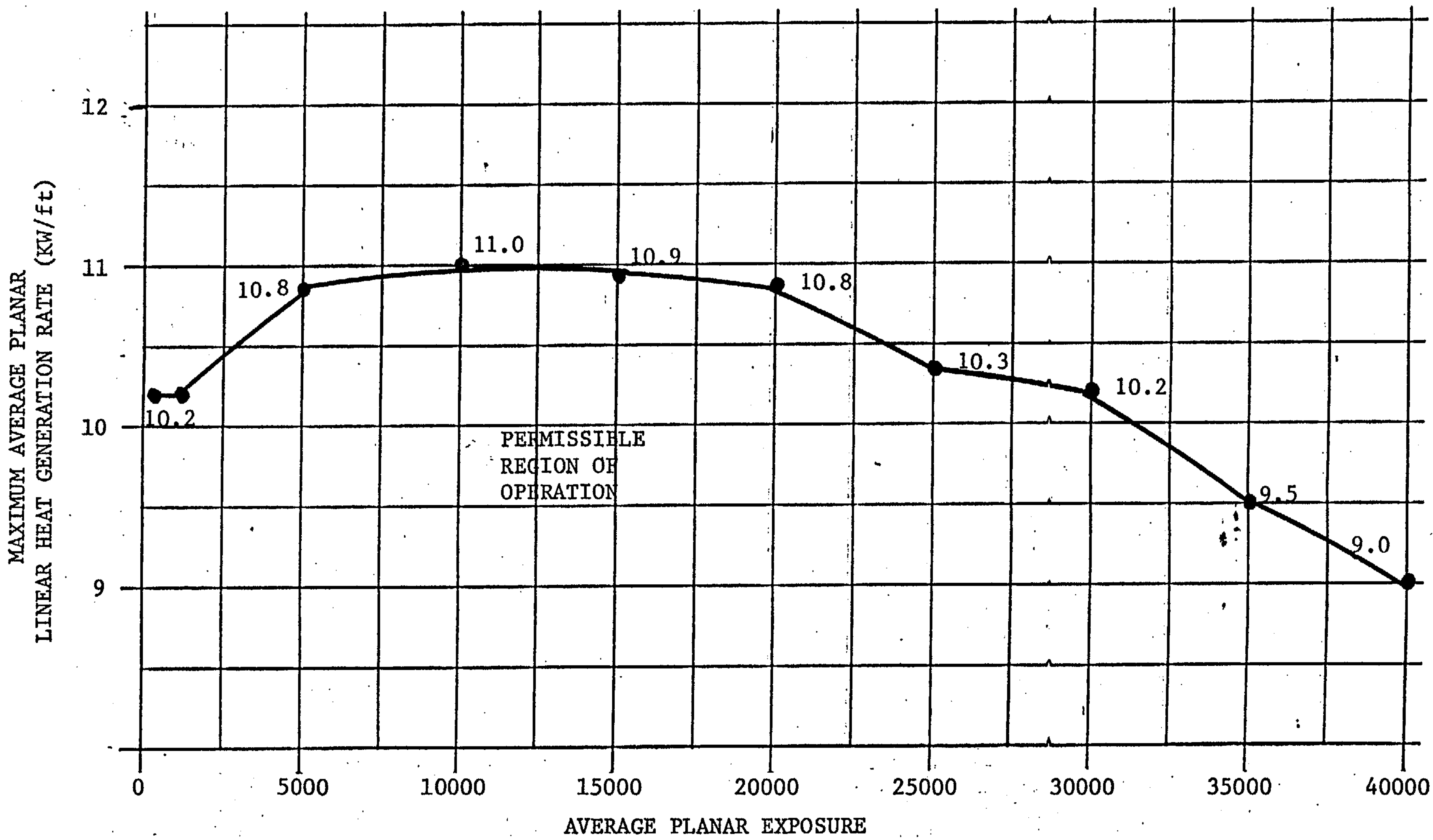
NOTE: See Footnote for Specification 3.2.1.



FUEL TYPE 8DRB265H
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE

FIGURE 3.2.1-6

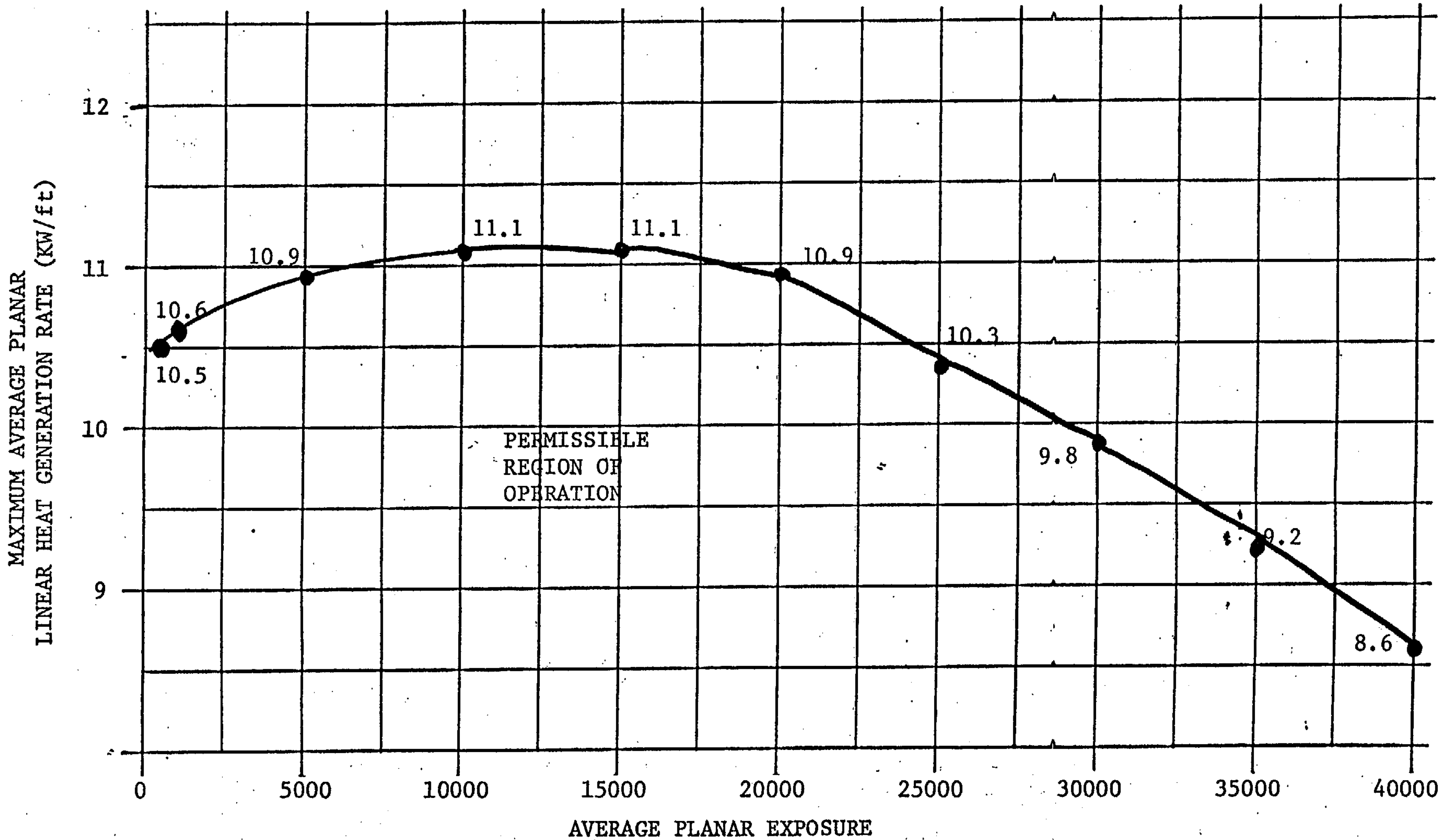
NOTE: See Footnote for Specification 3.2.1.



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

FIGURE 3.2.1-7

NOTE: SEE FOOTNOTE FOR SPECIFICATION 3.2.1.

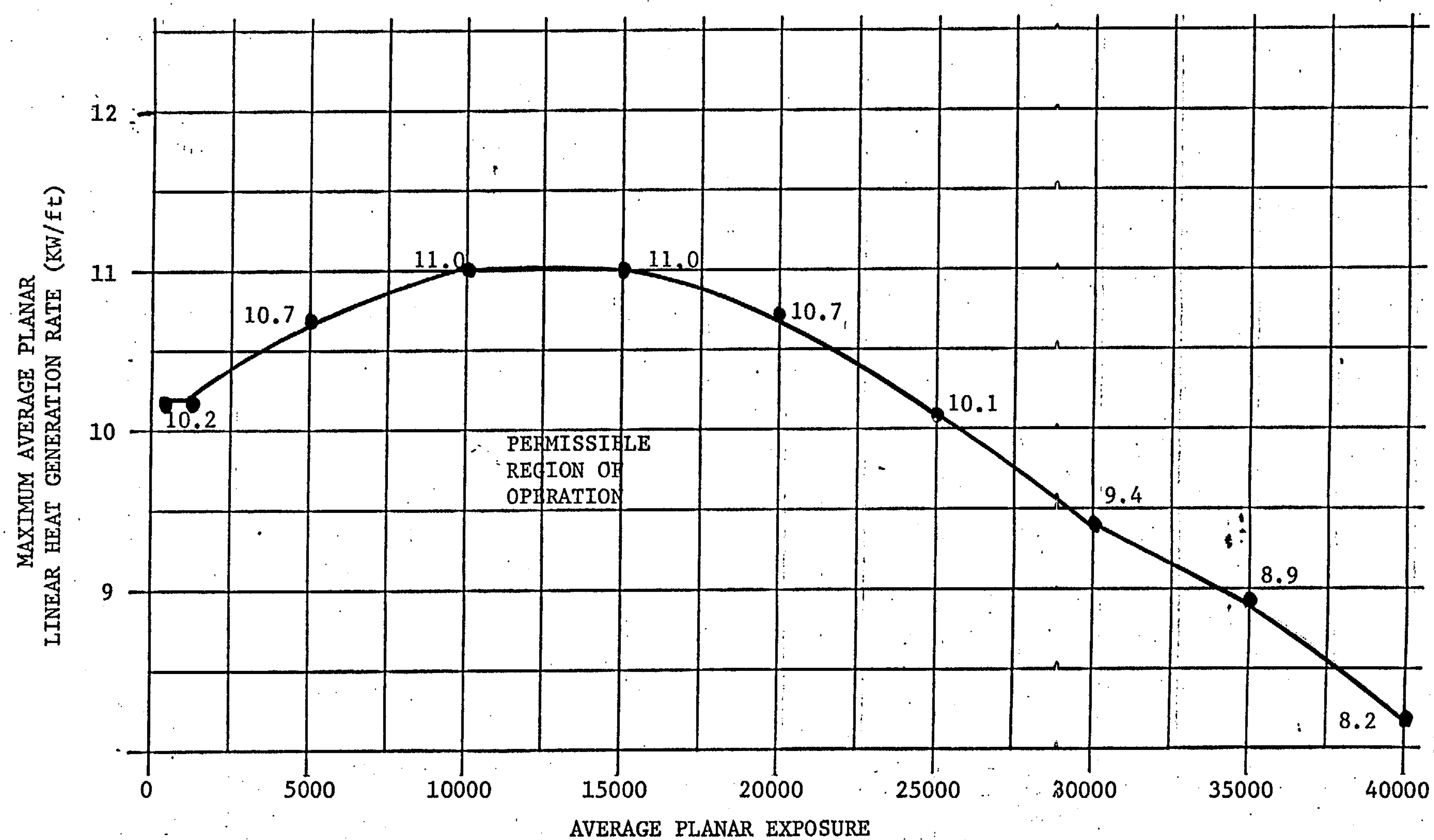


FUEL TYPE P8DRB265H

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

FIGURE 3.2.1-8

NOTE: SEE FOOTNOTE FOR SPECIFICATION 3.2.1.



FUEL TYPE P8DRB284 (P8x8R)

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

FIGURE 3.2.1-9

NOTE: SEE FOOTNOTE FOR SPECIFICATION 3.2.1.

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 percent of rated flow,
T = Lowest value of the ratio of design TPF divided by the MTPF obtained for any class of fuel in the core ($T \leq 1.0$), and

Design TPF for: P8 X 8R fuel = 2.39
8 X 8R fuel = 2.39
7 X 7 fuel = 2.60
8 X 8 fuel = 2.43

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 time limits within 4 hours, or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3.1 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, shall be equal to or greater than the MCPR limit times the K_f shown in Figure 3.2.3-1, provided that the end-of-cycle recirculation pump trip system is OPERABLE per specification 3.3.6.2, with:

a. If ODYN OPTION A analyses are in effect, the MCPR limits are listed below:

1. MCPR for 7x7 fuel = 1.21*
2. MCPR for 8x8 fuel = 1.29
3. MCPR for 8x8R fuel = 1.25 and
4. MCPR for P8x8R fuel = 1.27

b. If ODYN OPTION B analyses are in effect (refer to specification 3.2.3.2), the MCPR limits are listed below:

1. MCPR for 7x7 fuel = 1.20*
2. MCPR for 8x8 fuel = 1.29
3. MCPR for 8x8R fuel = 1.21 and
4. MCPR for P8x8R fuel = 1.22

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

ACTION:

a. With the end-of-cycle recirculation trip system INOPERABLE per Specification 3.3.6.2, operation may continue and the provisions of Specification 3.0.4 are not applicable with the following MCPR limit adjustments:

1. Beginning-of-cycle (BOC) to end-of-cycle (EOC) minus 2000 MWD/t, within one hour determine that MCPR, as a function of core flow, is equal to or greater than the MCPR limit times the K_f shown in Figure 3.2.3-1 with:

a. If ODYN OPTION A analyses are in effect, the MCPR limits are listed below:
1. MCPR for 7x7 fuel = 1.20*
2. MCPR for 8x8 fuel = 1.29,
3. MCPR for 8x8R fuel = 1.24 and,
4. MCPR for P8x8R fuel = 1.26

b. If ODYN OPTION B analyses are in effect (refer to specification 3.2.3.2), the MCPR limits are listed below:
1. MCPR for 7x7 fuel = 1.20*
2. MCPR for 8x8 fuel = 1.29
3. MCPR for 8x8R fuel = 1.22 and
4. MCPR for P8x8R fuel = 1.25

2. EOC minus 2000 MWD/t to EOC, within one hour determine that MCPR, as a function of core flow, is equal to or greater than the MCPR limit times the K_f shown in Figure 3.2.3-1 with:

~~a. If ODYN OPTION A analyses are in effect, the MCPR limits are listed below:~~
1. MCPR for 7x7 fuel = 1.28*
2. MCPR for 8x8 fuel = 1.35
3. MCPR for 8x8R fuel = 1.36 and
4. MCPR for P8x8R fuel = 1.39

b. If ODYN OPTION B analyses are in effect (refer to specification 3.2.3.2), the MCPR limits are listed below:
1. MCPR for 7x7 fuel = 1.20*
2. MCPR for 8x8 fuel = 1.29,
3. MCPR for 8x8R fuel = 1.24 and,
4. MCPR for P8x8R fuel = 1.27

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

SURVEILLANCE REQUIREMENTS

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

A. At least once per 24 hours,

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

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

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* For 7x7 fuel assemblies, the K_f factor is based on the 112% flow curve of Figure 3.2.3-1 rather than the actual setpoint of 102.5%.

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

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

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

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

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

i = Surveillance test number
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test
N_1 = Number of rods tested at BOC,
μ = 0.834 seconds
(mean value for statistical scram time distribution from de-energization of scram pilot valve solenoid to pickup on notch 36),
σ = 0.059 seconds
(standard deviation of the above statistical distribution).

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

ACTION:

Within twelve hours after determining that $\tau_{ave} > \tau_B$, the operating limit MCPRs shall be either:

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

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

where: τ_A = 0.900 seconds, control rod average scram insertion time limit to notch (36) per Specification 3.1.3.3,

$MCPR_{option\ A}$ = Determined from Table 3.2.3.2-1,

$MCPR_{option\ B}$ = Determined from Table 3.2.3.2-1, or,

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.2.3.2-1

TRANSIENT OPERATING LIMIT MCPR VALUES

TRANSIENT	FUEL TYPE 7x7		8x8		8x8R		P8x8R	
NONPRESSURIZATION TRANSIENTS								
With RPT operable (op.)	1.20		1.29		1.21		1.22	
With RPT inoperable (inop.)	1.20		1.29		1.22		1.25	
TURBINE TRIP/LOAD REJECT WITHOUT BYPASS								
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
RPT (op.)	1.21	1.13	1.25	1.17	1.25	1.17	1.27	1.19
RPT (inop.) BOC → EOC – 2000	1.18	1.08	1.23	1.08	1.24	1.08	1.26	1.08
RPT (inop.) EOC – 2000 → EOC	1.28	1.18	1.35	1.23	1.36	1.24	1.39	1.27
FEEDWATER CONTROL FAILURE								
	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$	$MCPR_A$	$MCPR_B$
RPT (op.)	1.16	1.13	1.17	1.14	1.17	1.14	1.17	1.14
RPT (inop.) BOC → EOC – 2000	1.16	1.10	1.16	1.10	1.17	1.11	1.17	1.11
RPT (inop.) EOC – 2000 → EOC	1.15	1.09	1.16	1.10	1.16	1.10	1.17	1.11

TABLE 3.3.4-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. APRM (C51-APRM-CH.A,B,C,D,E,F)		
a. Upscale (Flow-Biased)	$\leq (0.66\ W + 42\%)\ \frac{T^*}{MTPF}$	$\leq (0.66\ W + 42\%)\ \frac{T^*}{MTPF}$
b. Inoperative	NA	NA
c. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
d. Upscale (Fixed)	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 12% of RATED THERMAL POWER
2. ROD BLOCK MONITOR (C51-RBM-CH.A,B)		
a. Upscale	$\leq (0.66\ W + 39\%)\ \frac{T^*}{MTPF}$	$\leq (0.66\ W + 39\%)\ \frac{T^*}{MTPF}$
b. Inoperative	NA	NA
c. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale
3. SOURCE RANGE MONITORS (C51-SRM-K600A,B,C,D)		
a. Detector not full in	NA	NA
b. Upscale	$\leq 1 \times 10^5$ cps	$\leq 1 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3 cps	$\geq$ 3 cps
4. INTERMEDIATE RANGE MONITORS (C51-IRM-K601A,B,C,D,E,F,G,H)		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 of full scale	$\leq$ 108/125 of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3/125 of full scale	$\geq$ 3/125 of full scale

*T = 2.60 for 7 x 7 fuel.
T = 2.43 for 8 x 8 fuel.
T = 2.39 for 8 x 8R fuel.
T = 2.39 for P8 x 8R fuel.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The requirements of Specifications 3.9.1, and 3.9.3, and Table 1.2 may be suspended to permit the reactor mode switch to be locked in the Start-up position and to allow two control rods to be withdrawn for shutdown margin demonstrations provided at least the following requirements are satisfied:

a. The source range monitors are OPERABLE with the RPS circuitry shorting links removed per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, and

c. The "notch-override" control shall not be used during movement of the control rods.

APPLICABILITY: CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately restore the reactor mode switch to the Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 2 hours prior to the performance of a shutdown margin demonstration verify that:

a. The source range monitors are OPERABLE per Specification 3.9.2, and

b. The rod worth minimizer is OPERABLE with the required program, per Specification 3.1.4.1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity safety limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of 2.60 for 7 x 7 fuel, 2.43 for 8 x 8 fuel, 2.39 for 8 x 8R fuel and 2.39 for 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 2.60 for 7 x 7 fuel, 2.43 for 8 x 8 fuel, 2.39 for 8 x 8R and 2.39 for P8 x 8R fuel. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM high flux scram curve by the reciprocal of the APRM gain change. The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, 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 an instrument trip setting as given in Specification 2.2.1.

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

The limiting transient which determines the required steady state MCPR limit is the turbine trip with failure of the turbine bypass. This transient yields the largest Δ MCPR. 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 multichannel steady state flow distribution model as described in Section 4.4 of NEDO-20360[4] and on core parameters shown in Reference 3, response to Items 2 and 9.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 71 TO FACILITY LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-324

1.0 Introduction

By letter dated May 20, 1982 (Reference 1) the Carolina Power & Light Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit No. 2 (BSEP 2). The proposed changes revise the Technical Specifications to incorporate the limiting conditions for operation for the fifth fuel cycle (Cycle 5). By letters dated June 24, 1982 (Reference 2) and June 28, 1982 (Reference 3) the licensee modified the proposed Technical Specification changes. By letters dated June 7, June 21, July 1, and July 7, 1982 (References 4, 5, 6 and 7), the licensee submitted supplemental information relevent to this evaluation. References 3, 6, and 7 pertain to the core spray sparger crack discovered during the present BSEP 2 outage and to the required analyses and proposed corrective actions. The staff's evaluation of the sparger crack analyses and corrective actions is incorporated herein.

2.0 Evaluation

2.1 General

Generic information relative to the reload analysis of BWR fuel is presented in Reference 8. Reference 8 has been reviewed and previously approved by the staff (Reference 22). References 9 and 10 were submitted by the licensee to supplement the generic information with plant-specific information for BSEP 2, Cycle 5. For certain analyses, the licensee has used the ODYN code. We have approved the ODYN code (References 11 and 12) and have required its use in all BWR reload submittals received after February 1, 1981. For certain other analyses, the licensee has used the REDY code. We have previously reviewed and approved the REDY code in conjunction with the generic reload analysis.

Because we have previously reviewed a large number of generic considerations related to this type of core, and on the basis of evaluations presented in References 8 and 22, only a limited number of areas need to be addressed in this evaluation. For areas not addressed herein, the reader is referred to References 8 and 22.

2.2 Transient Analyses

2.2.1 Operating Limit Minimum Critical Power Ratio

The licensee has reviewed those transients that are the basis for Cycle 5 operation for BSEP 2 and has reanalyzed those transients that are critical with respect to safety margins and sensitive to the core reload parameter changes. The ODYN code was used in the determination of Critical Power Ratios (CPRs) for the rapid pressurization transients. The REDY code was used for the slower, non-pressurization events. The licensee provided both tabular and graphical results of its transient calculations for Cycle 5 in References 9 and 10 as supplemented by Reference 2. The most restrictive condition was calculated to occur as a result of a postulated generator load rejection without bypass. This is often the case for this type of BWR. For this event a Minimum CPR (MCPR) of 1.08 was predicted which is above the MCPR safety limit of 1.07.

Therefore, since the licensee, through the use of approved codes, has determined that the MCPR will remain above the safety limit MCPR for the most restrictive transient, we find the operating limit MCPR analysis to be acceptable.

2.2.2 Use of ODYN Option B MCPR Scram Insertion Time Conformance Procedure

The licensee has revised the operating MCPR Technical Specification to the ODYN "Option B" format where the Operating Limit MCPR (OLMCPR) varies with measured scram times. The specification is based on measurements to the notch 36 inserted position (20% scram time) which was chosen to coincide with present surveillance procedures at BSEP 2. The numerical values of the MCPRs are based on the CPRs from the load rejection without bypass transient given in Reference 1. The analysis of this transient was done using Technical Specification scram times but the uncertainty penalty applied to the nominal results was based on the results of a statistical analysis of transient response based on an improved scram insertion time distribution. The proposed Technical Specification changes include a verification that BSEP 2 is not outside this population distribution, or if it is, the OLMCPR linearly approaches a conservative value as predicted by ODYN option A analysis which incorporates a single penalty for uncertainties. Operation within the proposed limit will avoid violation of the Safety Limit MCPR at any time during Cycle 5.

We have reviewed the GE generic Option B scram time specification procedure using ODYN and have found it to be acceptable (References 11 and 12). Therefore, since the licensee is using an approved procedure, we find the BSEP 2 OLMCPR Technical Specification to be acceptable.

2.2 Transient Analyses

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The licensee has reviewed those transients that are the basis for Cycle 5 operation for BSEP 2 and has reanalyzed those transients that are critical with respect to safety margins and sensitive to the core reload parameter changes. The ODYN code was used in the determination of Critical Power Ratios (CPRs) for the rapid pressurization transients. The REDY code was used for the slower, non-pressurization events. The licensee provided both tabular and graphical results of its transient calculations for Cycle 5 in References 9 and 10 as supplemented by Reference 2. The most restrictive condition was calculated to occur as a result of a postulated generator load rejection without bypass. This is often the case for this type of BWR. For this event a Minimum CPR (MCPR) of 1.08 was predicted which is above the MCPR safety limit of 1.07.

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We have reviewed the GE generic Option B scram time specification procedure using ODYN and have found it to be acceptable (References 11 and 12). Therefore, since the licensee is using an approved procedure, we find the BSEP 2 OLMCPR Technical Specification to be acceptable.

2.2.3 Corrective Action for Reload MCPR Error

In Reference 1, the licensee provided a summary of a compensation for an error in the current version of the ODYN transient computer code which was used for performing the BSEP 2, Cycle 5, analysis of pressurization transients. The error has been identified by GE as a programming error in the axial power distribution calculation. The result of the correction is a predicted heat flux up to 2.1% less than the original version. In Reference 2 the licensee provided plant-specific results of a reanalysis of ODYN limiting transients using the corrected version of the ODYN code, and also provided modifications to the proposed Technical Specification changes. Transients analyzed by the REDY code are not affected by this change. We have reviewed the revised submittal that accounts for this error and find it to be acceptable and therefore, find the analysis to be acceptable.

2.2.4 Reactor Vessel Overpressure Protection

The licensee verified reactor vessel overpressure protection by an ODYN analysis of the closure of all main steam isolation valves (MSIV) with an indirect (flux) scram. MSIV closure is the limiting event with regard to vessel overpressurization. At the end of Cycle 5 with all safety relief valves operating and an indirect scram, the peak vessel pressure was predicted to be 1244 psig. Since this is below the peak allowable ASME overpressure of 1375 psig at the vessel bottom, we find the protection provided for reactor vessel overpressure to be acceptable.

2.3 Safety Limit Minimum Critical Power Ratio

The safety limit MCPR is established to assure that at least 99.9 percent of the fuel rods in the core do not experience boiling transition during the worst anticipated operational occurrence. The safety limit MCPR given for BSEP 2, Cycle 5 in References 1 and 13 is 1.07. This is consistent with Reference 8 and is, therefore, acceptable.

2.4 Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (Reference 14) show that the maximum thermal hydraulic stability decay ratio is 0.73 for BSEP 2, Cycle 5. This is typical for this type of BWR. This predicted maximum decay ratio is well below the 1.0 Ultimate Performance Limit decay ratio which we have previously found acceptable. In addition, BSEP 2 Technical Specifications prohibit operation in the natural circulation mode, thus providing a significant increase in reactor core stability operating margins. On the basis of the foregoing, we consider the thermal-hydraulic stability of BSEP 2, Cycle 5 to be acceptable.

2.5 Maximum Average Planar Linear Heat Generation Rate Limits (MAPLHGR)

2.5.1 MAPLHGR Endpoint Extension

In Reference 1, the licensee resubmitted the revised MAPLHGR curves for the 7 x 7 fuel types and also submitted revised MAPLHGR curves for the 8 x 8 fuel bundles in the proposed Cycle 5 core. The revision, which extends the MAPLHGR limits to 40,000 MWd/t for all fuel types, is based on methods (Reference 16) submitted as part of the application. Although the methodology used is generally applicable for these limits, we believe that the effects of enhanced fission gas release in high burnup fuel (above 20,000 MWd/MTU) were not adequately considered in the generic analysis. In response to this concern, GE requested (References 17 and 18) that credit for calculated peak cladding temperature margin as well as credit for approved, but unapplied, emergency core cooling system (ECCS) evaluation model changes be used to avoid MAPLHGR penalties at higher burnup. We found this proposal acceptable (Reference 19) provided that certain plant-specific conditions were met. In response to a generic letter (Reference 20) on the subject, the licensee stated (Reference 21) that it did not wish to apply the GE proposal to either Brunswick plant. However, in a followup letter (Reference 4) the licensee endorsed the GE position.

Therefore, since the licensee has endorsed the GE position with respect to enhanced fission gas release in high burnup fuel, and has used an otherwise acceptable generic analysis, we conclude that the MAPLHGR limits for BSEP 2 Cycle 5 are acceptable, except as noted in Section 2.5.3, below.

2.5.2 Effect of Core Spray Sparger Cracks on Emergency Core Cooling System (ECCS) Performance

As a result of the identification of a crack in the heat affected zone of one core spray sparger-to-junction box weld on BSEP 2 (see Section 2.6), the licensee evaluated the effects of a potential loss of spray distribution from the sparger This evaluation resulted in modification of the operating limit MAPLHGR valves by the application of a uniform 8.5 percent reduction of the values submitted in Reference 1. The new proposed Technical Specification values were provided in Reference 3.

At a meeting held in Bethesda on June 14-15, 1982, the licensee presented results of an analysis of the potential effects of the core spray sparger crack discovered on BSEP 2. The complete analysis was subsequently submitted under References 6 and 7. We conclude that no credit can be taken for core spray heat transfer from the affected sparger; thus necessitating a MAPLHGR reduction. The magnitude of the reduction was obtained by performing a bounding sensitivity calculation to determine the reduction required to meet the 10 CFR 50 Appendix K limit of 2200°F for peak clad temperature. The revised Loss of Coolant Accident (LOCA) calculations were performed using the standard approved analysis with the following assumptions: (1) no credit for upper plenum inventory, (2) no counter current flow limiting (CCFL) breakdown, and (3) no cooling contribution from the sparger with the crack before core reflooding. Assumptions (1) and (2) are conservative inputs to the currently approved Appendix K LOCA model and assumption (3) is appropriate for BSEP 2 with one core spray system inoperable. The analytical exception to Appendix K is that the spray heat transfer coefficient was set equal to zero. The limiting single failure (LPCI injection valve failure), break size, and location are unchanged from the original analysis for BSEP 2 Cycle 5. The resulting 8.5 percent reduction factor is independent of fuel exposure and is applied to the exposure dependent MAPLHGR values determined from calculations in which credit was taken for core spray heat transfer (provided in Reference 1).

Based on this review, we conclude that the discussion and LOCA calculations submitted for Cycle 5 operation resulting in reduced MAPLHGR values satisfy the criteria of 10 CFR 50.46 and are therefore acceptable.

2.6 Core Spray Sparger Crack

On May 20, 1982 the licensee advised Region II that it had identified a crack in the heat affected zone of one core spray sparger-to-junction box weld on BSEP 2. IE Bulletin 80-13 specifies, in part, that in the event such cracks are identified, licensees shall submit an analysis to NRR for review and approval prior to return to operation. The licensee presented the results of that analysis to the staff at meetings on June 14-15, 1982, and subsequently submitted the analysis under References 6 and 7. The LOCA reanalysis assessment is presented in Section 2.5, the structural integrity assessment is presented below.

The licensee's analysis concluded that no loadings have been identified which could result in stresses that would cause the sparger to break during normal plant operation, transients, or postulated LOCAs. The licensee's analysis also concluded that the possibility of crack propagation was extremely remote in this case since both crack ends extended into base metal. Thus, while the generally accepted and previously applied corrective actions included the installation of a clamp on the affected sparger, the licensee, because of the conclusions cited above, did not initially intend to install such a clamp. However, the licensee was subsequently in a position to, and did, install a clamp on the sparger.

We have determined that the licensee's conclusions and corrective actions in this matter are consistent with those for other similar situations for which continued operation was permitted. We thus conclude that operation of BSEP 2 with the subject sparger in its present configuration is acceptable.

3.0 Environmental Considerations

Based on the foregoing, 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, negative declaration or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: July 12, 1982

References

1. Letter, P. W. Howe (CPL) to D. B. Vassallo (NRC) dated May 20, 1982 (with attachments)

2. Letter, P. W. Howe (CPL) to D. B. Vassallo (NRC) dated June 24, 1982 (with attachments)

3. Letter, P. W. Howe (CPL) to D. B. Vassallo (NRC) dated June 28, 1982 (with attachments)

4. Letter, P. W. Howe (CPL) to D. B. Vassallo (NRC) dated June 7, 1982

5. Letter, S. R. Zimmerman (CPL) to D. B. Vassallo (NRC) dated June 21, 1982 (with attachments)

6. Letter, S. R. Zimmerman (CPL) to D. B. Vassallo (NRC) dated July 1, 1982 (with attachments)

7. Letter, S. R. Zimmerman (CPL) to D. B. Vassallo (NRC) dated July 7, 1982.

8. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-4, General Electric, January 1982

9. "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant, Unit 2, Reload 4," Y1003J01A37, March 1982 (without RPT)

10. Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant, Unit 2, Reload 4," Y1003J01A37, Revision 1, March 1982

11. "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors NEDO-24154 and NEDE-24154-P Volumes I, II, and III," June 1980

12. "Supplemental Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors NEDO-24154 and NEDE-24154P Volumes I, II, and III," January 1981

13. Letter, S. R. Zimmerman (CPL) to D. B. Vassallo (NRC) dated April 2, 1982

14. "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 2, Reload 4," X1003J01A37, DRF L12-00396-1, Class 1, General Electric, March 1982

15. N. C. Mosley (NRC/I&E) letter to J. P. O'Reilly (NRC/Region II) and others dated February 1, 1980

16. "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant, Unit 2, Reload 4," General Electric Company Report Y1003J01A37, Revision 1, March 1982

References (cont'd)

17. Letter, R. E. Engel (GE) to T. A. Ippolito (NRC) dated May 6, 1981

18. Letter, R. E. Engel (GE) to T. A. Ippolito (NRC) dated May 28, 1981

19. L. S. Rubenstein (NRC) memorandum for T. M. Novak (NRC) on "Extension of General Electric Emergency Core Cooling Systems Performance Limits" dated June 25, 1981

20. Letter, D. G. Eisenhut (NRC) to All Operating BWRs dated March 31, 1982

21. Letter, P. W. Howe (CPL) to D. B. Vassallo (NRC) dated May 19, 1982

22. Letter, T. A. Ippolito (NRC) to R. Gridley (GE) dated April 16, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-324

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. 71 to Facility Operating License No. DPR-62 issued to Carolina Power & Light Company (the licensee) which revised the Technical Specifications for operation of the 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 to establish revised safety and operating limits for operation of the facility during fuel Cycle Number 5.

The application for 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 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 Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated May 20, 1982, (2) supplemental submittals dated June 24, 1982 and June 28, 1982, (3) Amendment No. 71 to License No. DPR-62, and (4) 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. and at the Southport-Brunswick County Library, 109 West Moore Street, Southport, North Carolina 28461. A copy of items (3) and (4) 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 12th day of July 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing