June 1, 1987

Dockets Nos.: 50-321
and 50-366

Mr. James P. O'Reilly
Senior Vice President - Nuclear Operations
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. O'Reilly:

Subject: Issuance of Amendment Nos. 139 and 76 to Facility Operating Licenses DPR-57 and NPF-5 - Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TACS 64779/64780)

The Commission has issued the enclosed Amendments Nos. 139 and 76 to Facility Operating Licenses DPR-57 and NPF-5, for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated February 13, 1987 and supplemented May 18, 1987.

The amendments modify the Technical Specifications related to scram speed limit, scram speed measurement requirements, definition of design power, minimum critical power ratio limit, lead test fuel assemblies, and the average planar linear heat generation rate limits curve.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

Lawrence P. Crocker, Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II

Enclosures:
1. Amendment No. 139 to DPR-57
2. Amendment No. 76 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	PD#II-3/DRP-I/II
MDuncan/mac	LCrocker	BJYoungblood
05/22/87	05/22/87	05/ /87

Mr. J. P. O'Reilly
Georgia Power Company

Edwin I. Hatch Nuclear Plant,
Units Nos. 1 and 2
Docket Nos.: 50-366/321

cc:

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County Courthouse
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DATED June 1, 1987

AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE DPR-57, EDWIN I. HATCH, UNITS 1 & 2
AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NPF-05, EDWIN I. HATCH, UNITS 1 & 2

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
License No. DPR-57

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-57 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, (the licensee) dated February 13, 1987, and supplemented May 18, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   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-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 139, 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 its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

B. J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: June 1, 1987

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	OGC-Bethesda	PD#II-3/DRP-I/II
MDuncan/mac	LCrocker	M. Karman	BJYoungblood
05/12 /87	05/22/87	05/28/87	05/ /87

ATTACHMENT TO LICENSE AMENDMENT NO. 139

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Page	Insert Page
x	x
1.0-2	1.0-2
1.1-10	1.1-10
1.1-11	1.1-11
1.2-6	1.2-6
3.3-2	3.3-2
3.3-3	3.3-3
3.3-10	3.3-10
3.11-2	3.11-2
3.11-2a	3.11-2a
3.11-4a	3.11-4a
Figure 3.11-1 (Sheet 5)	Figure 3.11-1 (Sheet 5)
Figure 3.11-2 (Sheet 6)	Figure 3.11-2 (Sheet 6)
Figure 3.11-4	Figure 3.11-4

LIST OF FIGURES

C. <u>Core Alteration</u> - Core alteration shall be the addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls within the reactor pressure vessel 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 position.

D. <u>Design Power</u> - Design power refers to the power level at which the reactor is producing 105 percent of reactor vessel rated steam flow. Design power does not necessarily correspond to 105 percent of rated reactor power. The stated design power in megawatts thermal (MWt) is the result of a heat balance for a particular plant design. For Hatch Nuclear Plant Unit 1 the design power is approximately 2537 MWt.

E. <u>Engineered Safety Features</u> - Engineered safety features are those features provided for mitigating the consequences of postulated accidents, including for example containment, emergency core cooling, and standby gas treatment system.

F. <u>Hot Shutdown Condition</u> - Hot shutdown condition means reactor operation with the Mode Switch in the SHUTDOWN position, coolant temperature greater than 212°F, and no core alterations are permitted.

G. <u>Hot Standby Condition</u> - Hot standby condition means reactor operation with the Mode Switch in the START & HOT STANDBY position, coolant temperature greater than 212°F, reactor pressure less than 1045 psig, critical.

H. <u>Immediate</u> - Immediate means that the required action shall be initiated as soon as practicable, considering the safe operation of the Unit and the importance of the required action.

I. <u>Instrument Calibration</u> - An instrument calibration means the adjustment of an instrument output signal so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors.

J. <u>Instrument Channel</u> - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the HNP-1 Unit have been analyzed throughout the spectrum of planned operating conditions. The analyses were based upon plant operation in accordance with the operating map given in Figure 3-1 of Ref. 8. In addition, 2436 MWt is the licensed maximum power level of HNP-1, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. Active coolant flow is equal to 88% of total core flow. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted (see Figure 7-1, NEDO-21124-7) which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients, a MCPR equal to or greater than the actual operating limit MCPR is conservatively assumed to exist prior to initiation of the transients.

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various

2.1 FUEL CLADDING INTEGRITY (Continued)

power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

i. The licensed maximum power level is 2436 MWt.

ii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iii. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. IRM Flux Scram Trip Setting
The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in Figure 7.5-8 of the FSAR. Additional conserva-

2.2 REACTOR COOLANT SYSTEM INTEGRITY

A: Nuclear System Pressure

1. When Irradiated Fuel is in the Reactor

The 11 relief/safety valves are sized and set point pressures are established in accordance with the following requirements of Section III of the ASME Code:

a. The lowest relief/safety valve must be set to open at or below vessel design pressure and the highest relief/safety valve must be set to open at or below 105% of design pressure.

b. The valves must limit the reactor pressure to no more than 110% of design pressure.

The primary system relief/safety valves are sized to limit the primary system pressure, including transients, to the limits expressed in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels. No credit is taken from a scram initiated directly from the isolation event, or for power operated relief/safety valves, sprays, or other power operated pressure relieving devices. Thus, the probability of failure of the turbine-generator trip SCRAM or main steam isolation valve closure SCRAM is conservatively assumed to be unity. Credit is taken for subsequent indirect protection system action such as neutron flux SCRAM and reactor high pressure SCRAM, as allowed by the ASME Code. Credit is also taken for the dual relief/safety valves in their ASME Code qualified mode of safety operation. Sizing on this basis was applied to the most severe pressurization transient, which is the main steam isolation valves closure, starting from operation at 105 percent of the reactor warranted steamflow condition. The adequacy of this relief/safety valve sizing is verified each cycle by comparing the results of the analysis of the MSIV closure event starting from 102% of rated thermal power with the ASME limits described above.

Reference 2, Figure 4 shows peak, vessel bottom pressures attained when the main steam isolation valve closure transients are terminated by various modes of reactor scram, other than that which would be initiated directly from the isolation event (trip scram). Relief/safety valve capacities for this analysis are 84.0 percent, representative of the 11 relief/safety valves.

The relief/safety valve settings satisfy the Code requirements for relief/safety valves that the lowest valve set point be at or below the vessel design pressure of 1250 psig. These settings are also sufficiently above the normal operating pressure range to prevent unnecessary cycling caused by minor transients. The results of postulated transients where inherent relief/safety valve actuation is required are given in Section 14.3 of the FSAR.

2. When Operating the RHR System in the Shutdown Cooling Mode

An interlock exists in the logic for the RHR shutdown cooling valves, which are normally closed during power operation, to prevent opening of the valves above a preset pressure setpoint of 145 psig. This setpoint is selected to assure that pressure integrity of the RHR system is maintained. Administrative operating procedures require the operator to

3.3.B.2. Excessive Scram Time

Control rods with a scram insertion time to reach notch position 6 which exceeds 7.00 seconds shall be considered inoperable, but if they can be moved with control rod drive pressure, they need not be fully inserted or disarmed electrically.

3.3.B.3. Inoperable Accumulators

Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

4. Limiting Number of Inoperable Control Rods

During reactor power operation, no more than one control rod in any 5 x 5 array may be inoperable (at least 4 operable control rods must separate any 2 inoperable ones). If this Specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a shutdown condition within 24 hours.

C. Control Rod Drive System

1. Control Rod Drive Coupling Integrity

Each control rod shall be coupled to its drive or completely inserted and its directional control valves disarmed electrically except during control rod drive maintenance as stated in Specification 3.10.E.

4.3.B. Operable Control Rod Exercise Requirements (Cont'd)

When it is initially determined that a control rod is incapable of normal insertion, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted the reactor shall be brought to the Cold Shutdown Condition within 24 hours and a shutdown margin test made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted.

Once per week, check the status of the pressure and level alarm for each accumulator.

4.3.C. Control Rod Drive System

1. Control Rod Drive Coupling Integrity

The coupling integrity shall be verified for each withdrawn control rod as follows:

a. When the rod is withdrawn the first time after each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation and rod position indication including where applicable the "full-in" and "full-out" position. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 30% power shall be performed to verify instrumentation response.

3.3.C.2. Scram Insertion Times

a. All Operable Control Rods

The average scram insertion time of all operable control rods at a reactor dome pressure $\geq$ 950 psig based on the de-energization of the scram pilot valve solenoids as time zero, shall be no greater than:

Notch Position From Fully Withdrawn	Average Scram Insertion Time (Sec)
46	0.358
36	1.096
26	1.860
5	3.419

b. Three Out of Four Rods in a Two-by-Two Array

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array at a reactor dome pressure $\geq$ 950 psig shall be no greater than:

Notch Position From Fully Withdrawn	Average Scram Insertion Time (Sec)
46	0.379
36	1.162
26	1.972
6	3.624

4.3.C.1.b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

4.3.C.2. Scram Insertion Times

a. After each refueling outage all control rods capable of normal insertion shall be scram time tested from the fully withdrawn position after a reactor dome pressure of 950 psig has been attained. This testing must be complete before 40% rated thermal power is exceeded.

b. Routine Time Tests

At 16-week intervals, 10% of the control rods capable of movement with control rod drive pressure shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3.3.C. Control Rod Drive System

1. Control Rod Drive Coupling Integrity

Limiting Conditions for Operation:

Operability of the control rod drive system requires that the drive be coupled to the control rod. In the analysis of control rod drop accidents it has been assumed that one control rod drive coupling has lost its integrity. To assure that not more than one coupling could be in this condition, it is required that either a drive is coupled to the control rod or the drive is fully inserted and disarmed electrically. This requirement serves to maintain operation within the envelope of conditions by the plant safety analyses.

Surveillance Requirements:

Observation of a response from the nuclear instrumentation during an attempt to withdraw a control rod provides an indication that the rod is following the drive. The overtravel position feature provides a positive check on the coupling integrity, for only an uncoupled drive can reach the overtravel position.

2. Scram Insertion Times

Limiting Conditions for Operation:

The control rod drive system is designed to bring the reactor subcritical at a rate fast enough to prevent excessive fuel damage. Analysis of the limiting transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specification provide the required protection and MCPR remains greater than 1.07. The limit on the number and pattern of rods permitted to have long scram times is specified to assure that the effect of rods of long scram times are minimized in regard to reactivity insertion rate. Grouping of long scram time rods is prevented by not permitting more than one slow rod in any four rod array. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no operable control rod to have a scram insertion time to notch position 06 greater than 7 seconds.

3.11.B. Linear Heat Generation Rate (LHGR) (Continued)

operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

C. Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR), which is a function of scram time, core power, and core flow. For 25% $\leq$ power < 30%, the OLMCPR is given in Figure 3.11.6. For power $\geq$ 30%, the OLMCPR is the greater of either:

1. The applicable limit determined from Figure 3.11.3, or
2. The applicable limit from either Figures 3.11.4 or 3.11.5 multiplied by the K_p factor determined from Figure 3.11.6, where τ is the relative measured scram speed with respect to Option A and Option B scram speeds. If τ is determined to be less than zero, then the OLMCPR is evaluated at $\tau = 0$.

4.11.C.1. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined to be equal to or greater than the applicable limit, 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.F.

4.11.C.2. Minimum Critical Power Ratio Limit

The MCPR limit at rated flow and rated power shall be determined for each fuel type, as appropriate, from figure 3.11.4 or 3.11.5 using:

a. $\tau = 1.0$ prior to initial scram time measurements for the cycle, performed in accordance with specifications 4.3.C.2.a.

or

b. τ is determined from scram time measurements performed in accordance with specification 4.3.C.2.

The determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by specification 4.3.C.2.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.11.C. Minimum Critical Power Ratio (MCPR)

If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four(4) hours. If the Limiting Condition for Operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

D. Reporting Requirements

If any of the limiting values identified in Specifications 3.11.A., B., or C. are exceeded, a Reportable Occurrence report shall be submitted.

If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this specification.

3.11.C. Minimum Critical Power Ratio (MCPR) (Continued)

According to Figure 3.11.4 or 3.11.5, the 100% power, 100% flow operating limit MCPR (OLMCPR) depends on the average scram time, τ, of the control rods, where:

$$\tau = 0 \text{ or } \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}, \text{ whichever is greater}$$

where: τ_A = 1.096 sec (Specification 3.3.C.2.a, scram time limit to notch 36)

$$\tau_B = \mu + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i} \right]^{1/2} \sigma \quad \text{[Reference 10]}$$

where: μ = 0.822 sec (mean scram time used in the transient analysis)

σ = .018 sec (standard deviation of μ)

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

where: n = number of surveillance tests performed to date in the cycle

N_i = number of active control rods measured in the ith surveillance test

τ_i = average scram time to notch 36 of all rods in the ith surveillance test

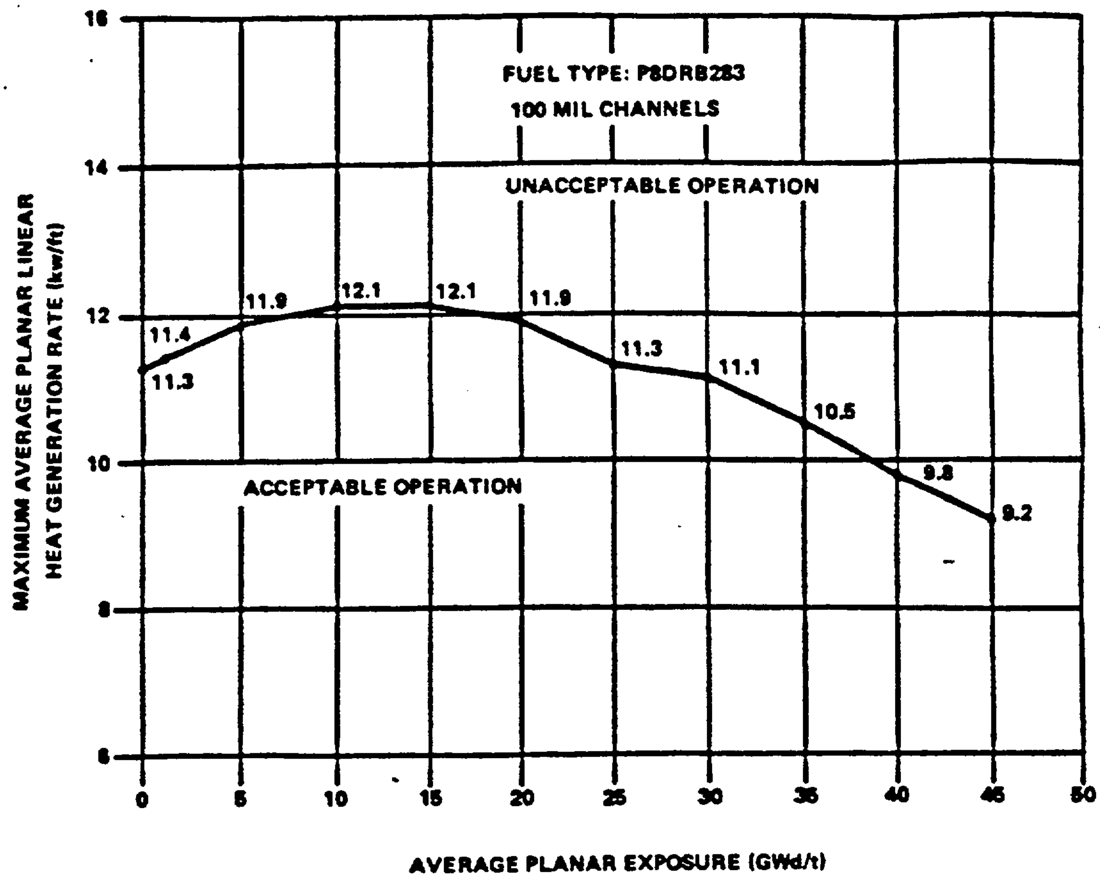
N_1 = total number of active rods measured in 4.3.C.2.a

The purpose of the $MCPR_f$, and the K_p of Figures 3.11-3 and 3.11-6, respectively, is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power, the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

The core power dependent MCPR operating limit $MCPR_p$ is the power rated flow MCPR operating limit multiplied by the K_p factor given in Figure 3.11-6.

The K_ps are established to protect the core from transients other than core flow increases, including the localized event such as rod withdrawal error. The K_ps were determined based upon the most limiting transient at the given core power level. (For further information on MCPR operating limits for off-rated conditions, reference NEDC-30474-P.(11))



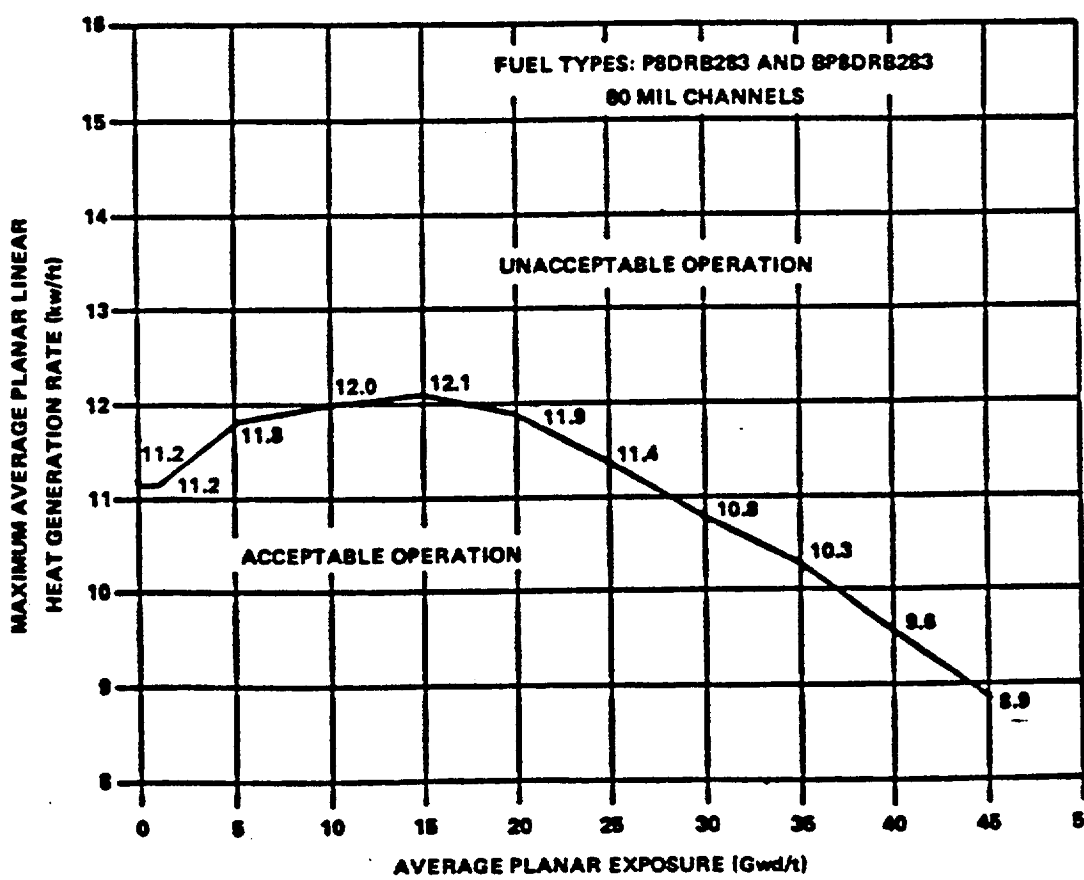
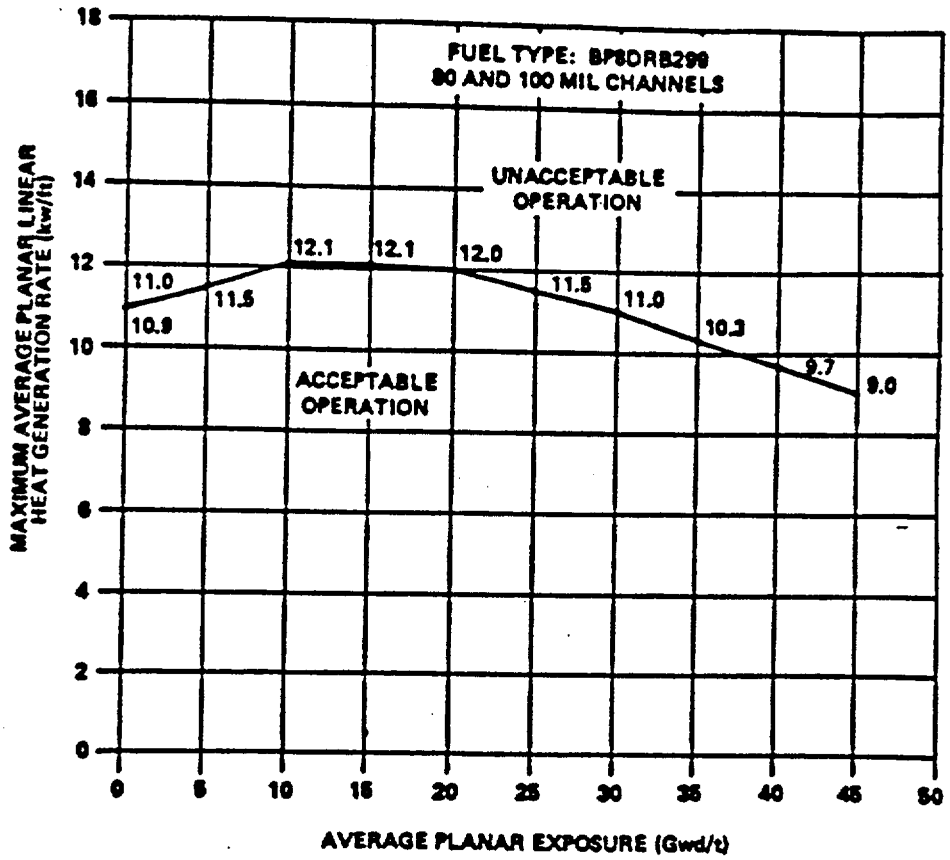


FIGURE 3.11-1 (SHEET 5)



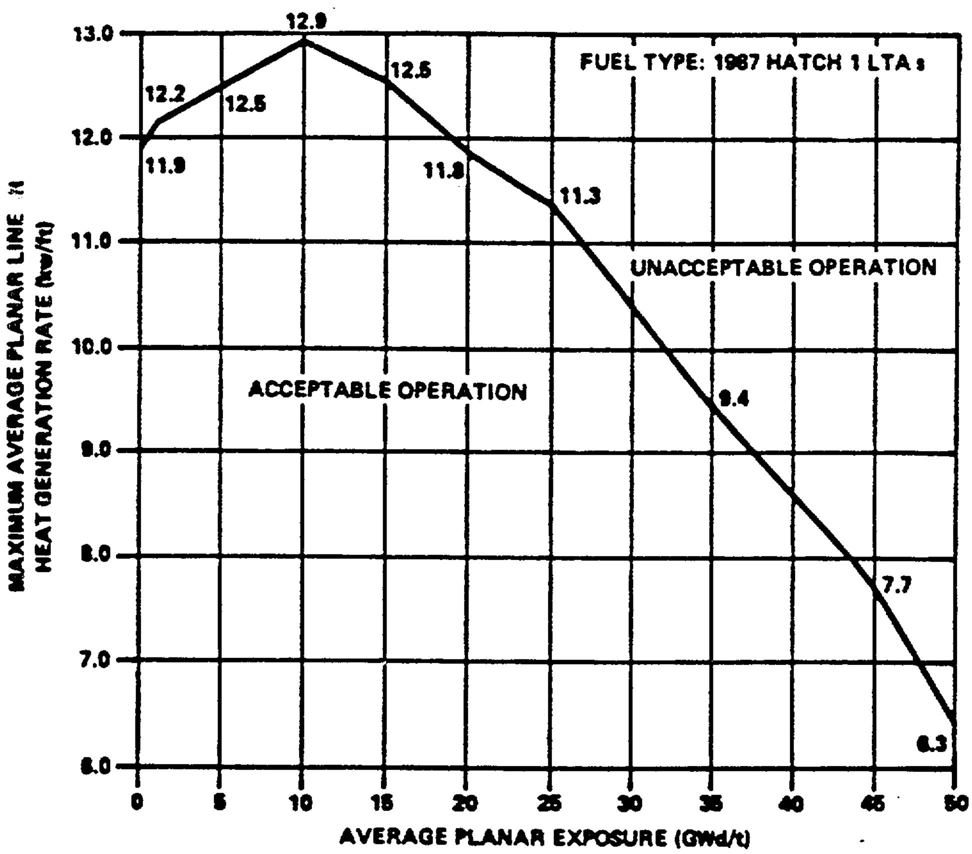


FIGURE 3.11-1 (SHEET 6)

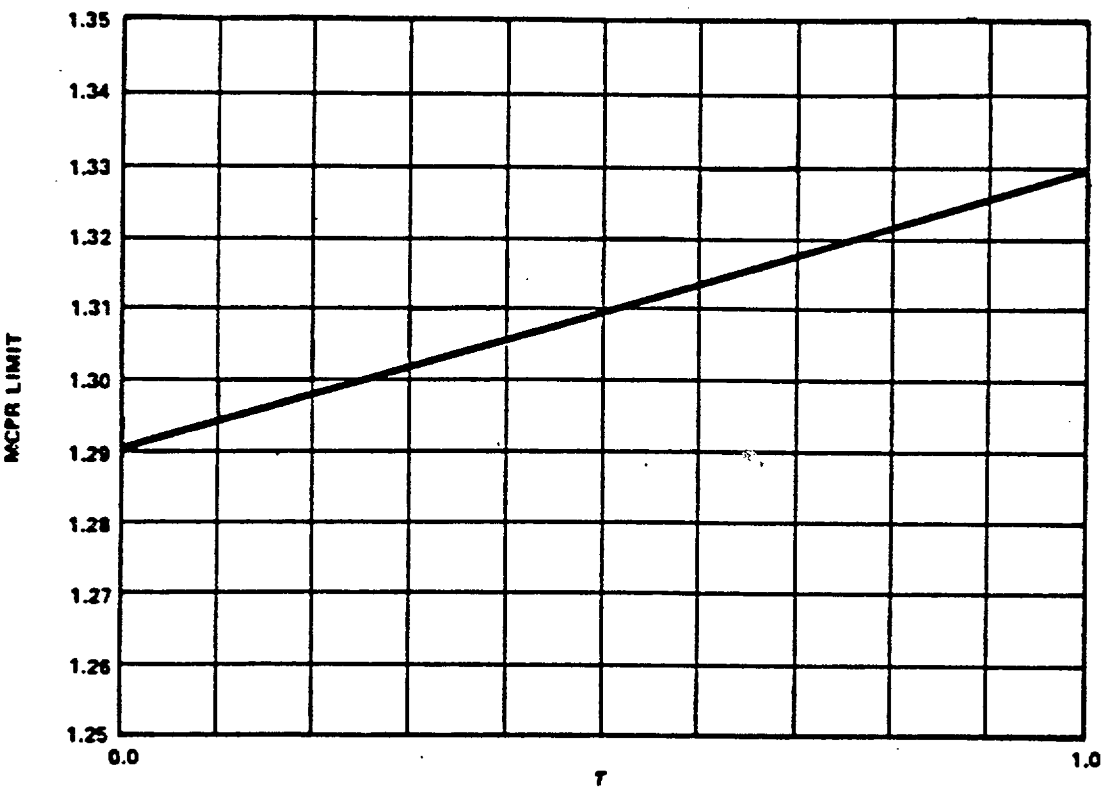


FIGURE 3.11-4
MCPR LIMIT FOR ALL 8X8 FUEL TYPES
FOR RATED POWER AND RATED FLOW

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, (the licensee) dated February 13, 1987, as supplemented May 18, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   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. NPF-5 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76, 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 its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

B. J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 1, 1987

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	OGC-Bethesda	PD#II-3/DRP-I/II
MDuncan/mac	LCrocker	M. Karman	BJYoungblood
05/12/87	05/22/87	05/28/87	05/ /87

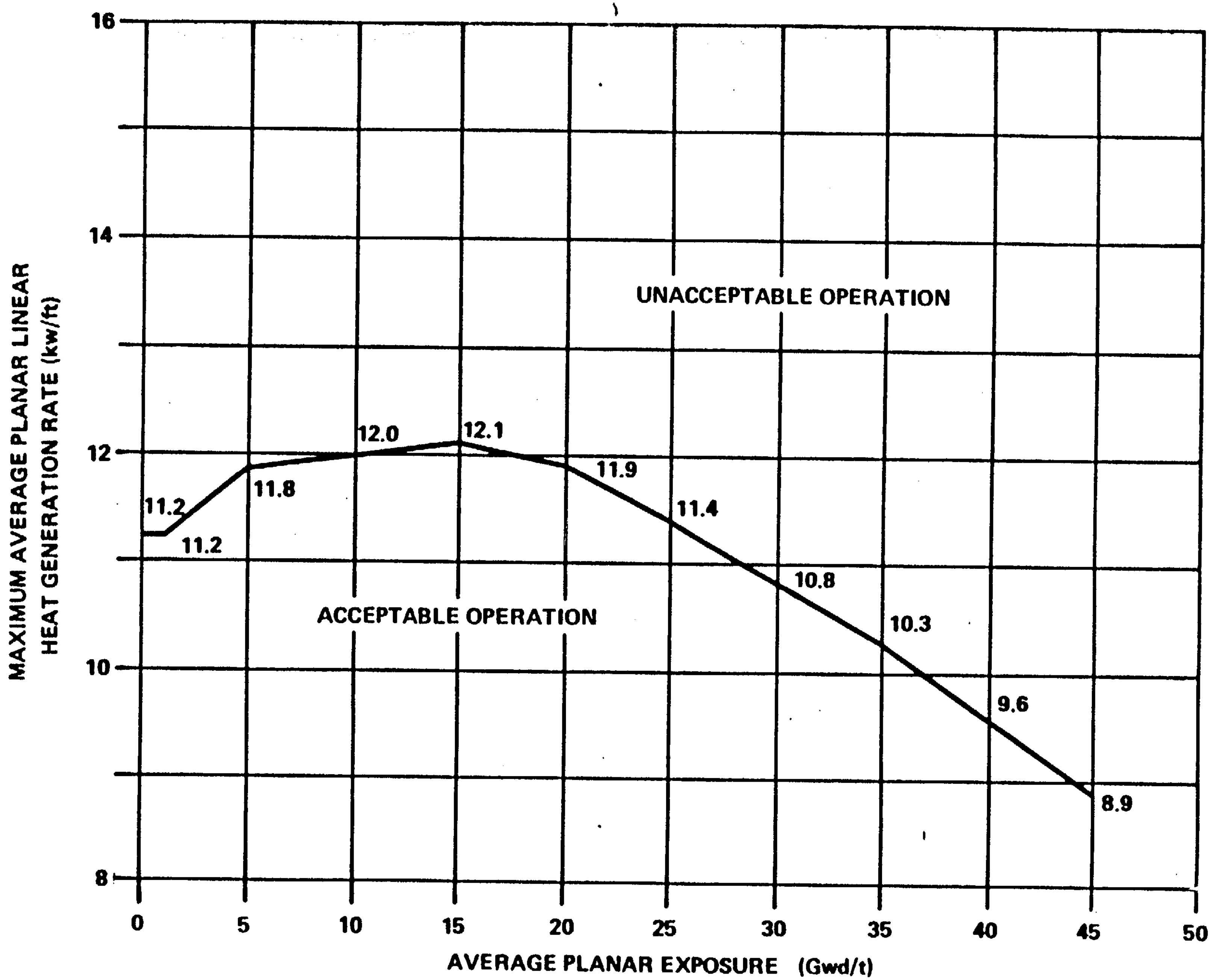
ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Page	Insert Page
3/4 2-4h	3/4 2-4h



FUEL TYPES P8DRB283 AND BP8DRB283 80 MIL CHANNELS
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-10

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS.139 AND 76 TO

FACILITY OPERATING LICENSES DPR-57 AND NPF-5

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

INTRODUCTION

By letter dated February 13, 1987 (Reference 1), supplemented by letter dated May 18, 1987 (Reference 2), Georgia Power Company (the licensee) proposed changes to the Plant Hatch Units 1 and 2 Technical Specifications that would: (1) revise the Minimum Critical Power Ratio (MCPR) scram time parameters for both Units 1 and 2; (2) replace the current method of measuring control rod scram times for Unit 1 by the method currently used for Unit 2; (3) revise the initial power assumed for certain transients for Unit 1; (4) reduce the Option A MCPR limit for Unit 1 from 1.35 to 1.33; (5) add an Average Planar Linear Heat Generation Rate (APLHGR) limit curve to reflect the thermal-mechanical and Emergency Core Cooling System (ECCS) limits on four Lead Test Assemblies (LTAs) which are expected to be part of the Reload 10 fuel batch for Unit 1; and (6) modify the APLHGR limit curves for P8 DRB 283 and BP8 DRB 283 80-mil fuel for both Unit 1 and Unit 2 to include a previously omitted data point at 1.0 Gwd/t.

EVALUATION

The proposed Technical Specification (TS) changes fall into six categories, and are discussed individually.

(1) Revise the MCPR scram time parameters.

This proposed change has to do with the Option A and Option B scram speed formulation. The licensee's February 13, 1987 letter (Reference 1) proposed to change the constants used in determining the ODYN Code option B scram speed limit for both Unit 1 and Unit 2. The change would be from the constants based on the GENESIS set of methods to those approved by the staff from a larger data base for the GEMINI methods. This is an acceptable change. In making the change the licensee proposed to remove from the TS the formula, definitions and associated constants for the various scram times used in determining MCPR limits via ODTN option A and B methods in TS 3/4.11.C.2 for Hatch 1 and TS 3/4.2.3 in Hatch 2. They would retain the information in the plant procedures. The staff found this acceptable provided the information removed from the TS also was included in the appropriate Bases, since it provided information relevant to scram speed and there is more than one candidate for the constants. This was discussed with the licensee and, by letter dated May 18, 1987 (Reference 2), the licensee requested that the Unit 1 Bases be modified to include the information that would be deleted from the TS.

In the same letter (Reference 2) the licensee stated that the current Unit 2 operating cycle was calculated using the older GENESIS methodology and that it would be technically correct to leave the information relating to this methodology in the Unit 2 TS for now. At a later date, a change to the Unit 2 methodology will be requested such that next operating cycle of Unit 2 can be calculated using the newer GEMINI methods. Accordingly, the licensee's May 18, 1987, letter withdraws the request to change the Unit 2 TS at this time.

This change, therefore, applies only to Unit 1 instead of to both Unit 1 and Unit 2 as stated in the Federal Register notice of this action (52 FR 9570, March 25, 1987).

Since there, thus, is no change to the Unit 2 TS and the change to the Unit 1 TS is acceptable, we conclude that the overall change to scram time parameters is acceptable.

(2) Change the method of measuring control rod scram times for Unit 1.

Scram time requirements for Unit 1 are currently specified (TS 3.3.B.2 and 3.3.C.2 and Bases 3.3.C.2) in terms of percent insertion versus time. The licensee proposes to convert the specifications from percent insertion versus time to notch position versus time. The notch position is directly related to actual control rod insertion, whereas the percent insertion requires a conversion calculation. The proposed new times specified by notch position are exactly equivalent to the existing scram speed requirements, so there is no fundamental change in the specifications. The notch system is used on Hatch Unit 2 and is in line with the Standard TS. It is, therefore, acceptable.

(3) Revise the initial power assumed for certain transients for Unit 1.

The licensee proposes to change the wording in the Unit 1 TS Bases 2.1 and 2.2 and in the definitions of Design Power. The initial power level assumed for some transient analyses would be changed to be consistent with the power level used in the General Electric Company (GE) GEMINI methodology (Reference 3), as approved by the staff (Reference 4). In GEMINI, the power level uncertainty is included in adjustment factors and events are analyzed at rated rather than design power. The proposed changes reflect the differences in calculation methods and are acceptable.

(4) Reduce the Option A MCPR limit for Unit 1.

TS Figure 3.11-4 presents a curve of MCPR limits as a function of average measured scram speed (ODYN Code, Option A and B) for all 8 x 8 fuel types. The licensee proposes to reduce the Option A MCPR limit from 1.35 to 1.33 (thereby changing the curve in Figure 3.11-4) as a result of a change from the GENISIS to the GEMINI methods and uncertainty analysis. The GEMINI method is approved by the staff and the proposed change in MCPR limit for Option A scram speed is reasonable and acceptable.

(5) Add an Average Planar Linear Heat Generation Rate (APLHGR) limit curve to the Unit 1 TS to reflect the thermal-mechanical and ECCS limits on four Lead Test Assemblies (LTAs) which are expected to be part of the Reload 10 fuel batch for Unit 1.

The licensee proposes to add to Unit 1 TS Figure 3.11-1 (Sheet 6) a new curve showing the maximum APLHGR (MAPLHGR) as a function of burnup for the LTA fuel type expected to be inserted (four assemblies) in the Reload 10 fuel batch. The fuel assembly is similar to LTA fuel recently approved by the staff for insertion (in limited amounts) in other reactors, e.g., Peach Bottom 2, Cycle 8. It is described in reports by GE, included as part of the Hatch submittal. These reports also describe the standard GESTAR II (NEDE-24011-P-A-8, "General Electric Standard Application for Reactor Fuel") methods which will be used to analyze the transients and accidents required to demonstrate that the LTA will operate within all design and safety limits. The LOCA analyses for the LTA in Hatch Unit 1, which provided the MAPLHGR values, were done on an assumed 14.4 Kw/ft LHGR limit, as has been approved for other reactors for this type of LTA fuel, but the operating limit in Hatch Unit 1 will remain at 13.4 Kw/ft; thus, no TS change for LHGR is needed for the LTA in Hatch Unit 1. Also, unlike more complex TS provided for some previously approved similar LTA fuel, a single MAPLHGR curve will apply to all axial regions of the assembly. The analyses providing the LTA MAPLHGR values and the resulting proposed curve, and the descriptions, fuel analyses, and proposed analyses for operation with the fuel in Hatch Unit 1 are acceptable.

(6) Modify the APLHGR limit curve for P8 DRB 283 and BP8 DRB 283 80-mil fuel for both Unit 1 and Unit 2 to include a previously omitted data point at 1.0 Gwd/t.

The licensee proposes to add a MAPLHGR data point at an exposure of 1.0 Gwd/t to the curves for two existing fuel types, P8 DRB 283 and BP8 DRB 283 with 80-mil channels, (Figure 3.11-1 (Sheet 5) for Hatch Unit 1 and Figure 3.2.1-10 for Hatch Unit 2), which had been inadvertently omitted. These are data points from the original GE analysis and their addition to the TS figures is acceptable.

We have reviewed the information submitted for proposed TS changes for Hatch 1 and 2 relating to scram speed, power level definition and MCPR and MAPLHGR limits. Based on this review we conclude that appropriate material was submitted and the changes are reasonable and acceptable.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 9570) on March 25, 1987, and consulted with the state of Georgia. No public comments were received, and the state of Georgia did not have any comments.

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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Letter from James P. O'Reilly, Georgia Power Company, to the NRC, dated February 13, 1987.

2. Letter from L.T. Gucwa, Georgia Power Company, to the NRC, dated May 18, 1987.

3. Letter, J. S. Charnley (GE) to H. N. Berkow (NRC, "Revised Supplementary Information Regarding Amendment 11 to GE Licensing Topical Report NEDE-24011-P-A" (GESTAR II), dated January 16, 1986.

4. Letter from G. C. Lainas (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, "General Electric Licensing Reload Report', Supplement to Amendment 11," dated March 22, 1986.

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Dated: June 1, 1987