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Docket No. 50-366

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Mr. J. T. Beckham, Jr.
Vice President -Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

The Commission has issued the enclosed Amendment No. 28 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 11, 1982.

This amendment revises the TSs to enable operation of the Plant after the Cycle 3 Reload, at licensed power.

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

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 28 to NPF-5
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Concur to form of amend & Notice only; on condition "TSs" changed to "Technical Specifications" in public Notice

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NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960 cc w/enclosure(s):

G. F. Trowbridge, Esq.
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

Ruble A. Thomas
Vice President
P. O. Box 2625
Southern Company Services, Inc.
Birmingham, Alabama 35202

Ozen Batum
Southern Company Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202

Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

Mr. L. T. Gucwa
Georgia Power Company
Engineering Department
P. O. Box 4545
Atlanta, Georgia 30302

Mr. Max Manry
Georgia Power Company
Edwin I. Hatch Plant
P. O. Box 442
Baxley, Georgia 31513

Regional Radiation Representative
EPA Region IV
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Appling County Public Library
301 City Hall Drive
Baxley, Georgia 31513

Mr. R. F. Rogers
U. S. Nuclear Regulatory Commission
Route 1, P. O. Box 279
Baxley, Georgia 31513

Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

cc w/enclosure(s) & incoming dtd.:
3/11/82

Charles H. Badger
Office of Planning and Budget
Room 610
270 Washington Street, S.W.
Atlanta, Georgia 30334

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. 28
License No. NPF-5

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated March 11, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. 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. 28, 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

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 6, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 28

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 area of change. The overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 2-1	3/4 2-1
--	3/4 2-4B
3/4 2-7a	3/4 2-7a
B3/4 2-1	B3/4 2-1
3/4 3-40	3/4 3-40

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: CONDITION 1, when THERMAL POWER $\geq$ 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, or 3.2.1-5, initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

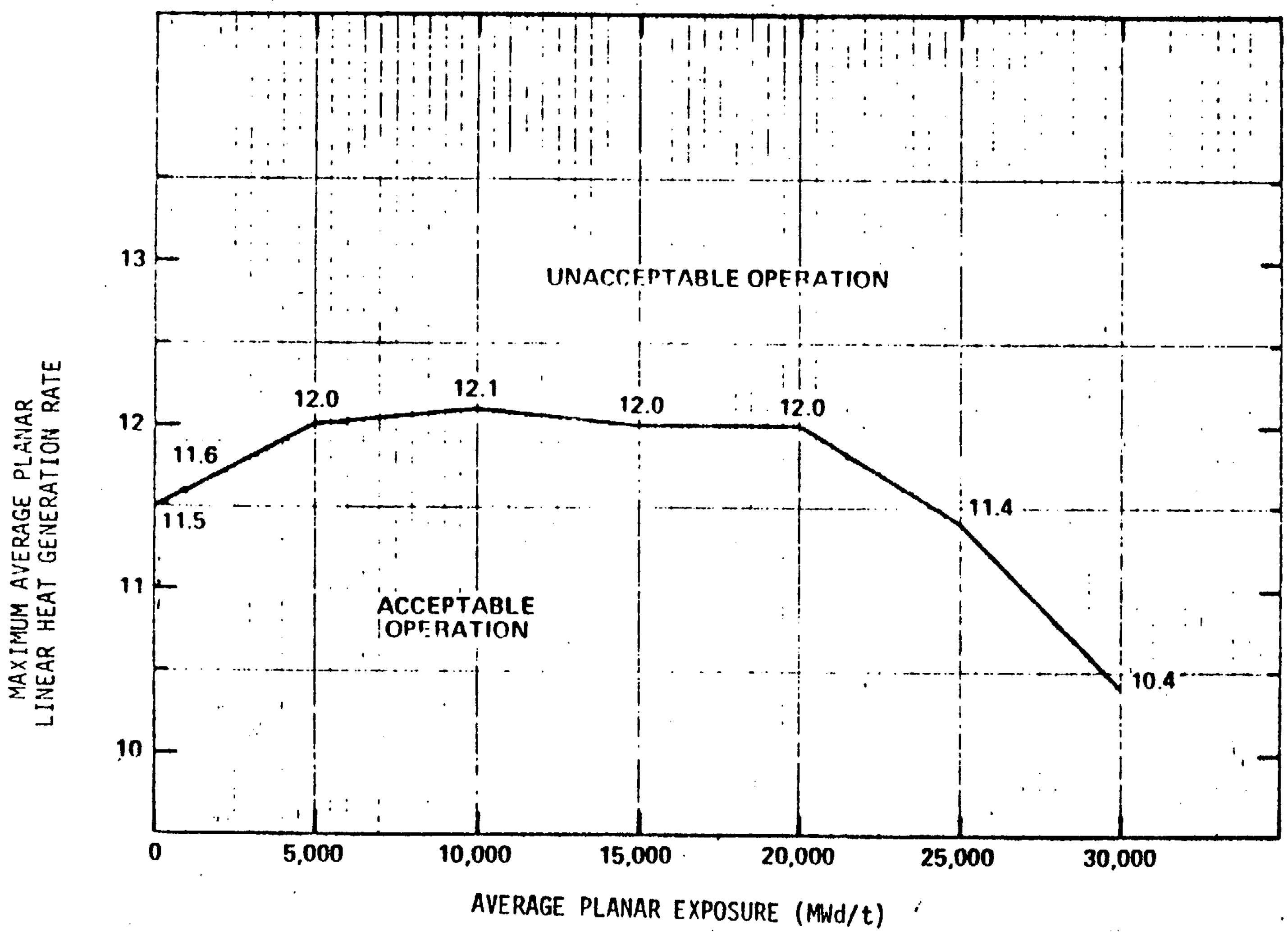
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the applicable limit determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, or 3.2.1-5:

a. At least once per 24 hours,

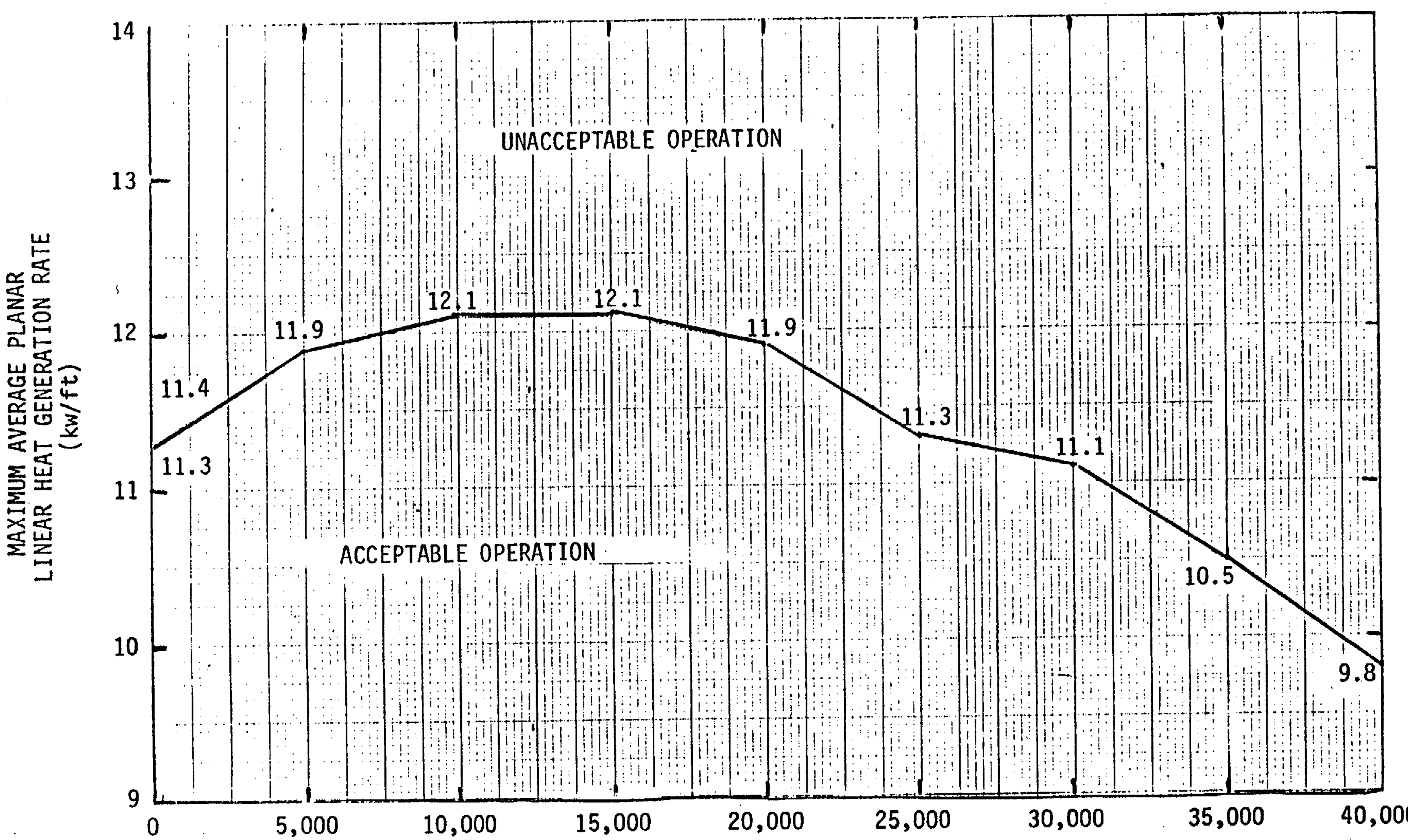
b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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



FUEL TYPE 8DR 183

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-1



AVERAGE PLANAR EXPOSURE (Mwd/st)

FUEL TYPE P8DRB283

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-5

3/4.2.3 MINIMUM CRITICAL POWER RATIO

SURVEILLANCE REQUIREMENTS (Continued)

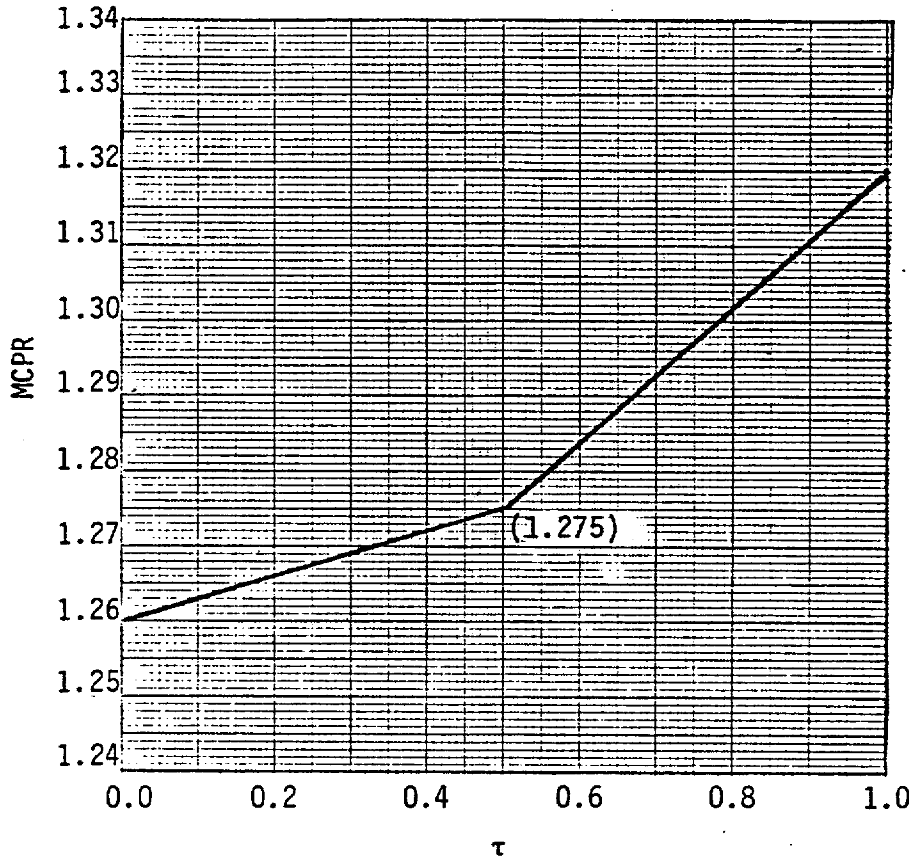
b. τ as defined in Specification 3.2.3; the determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2.

MCPR shall be determined to be equal to or greater than the applicable limit:

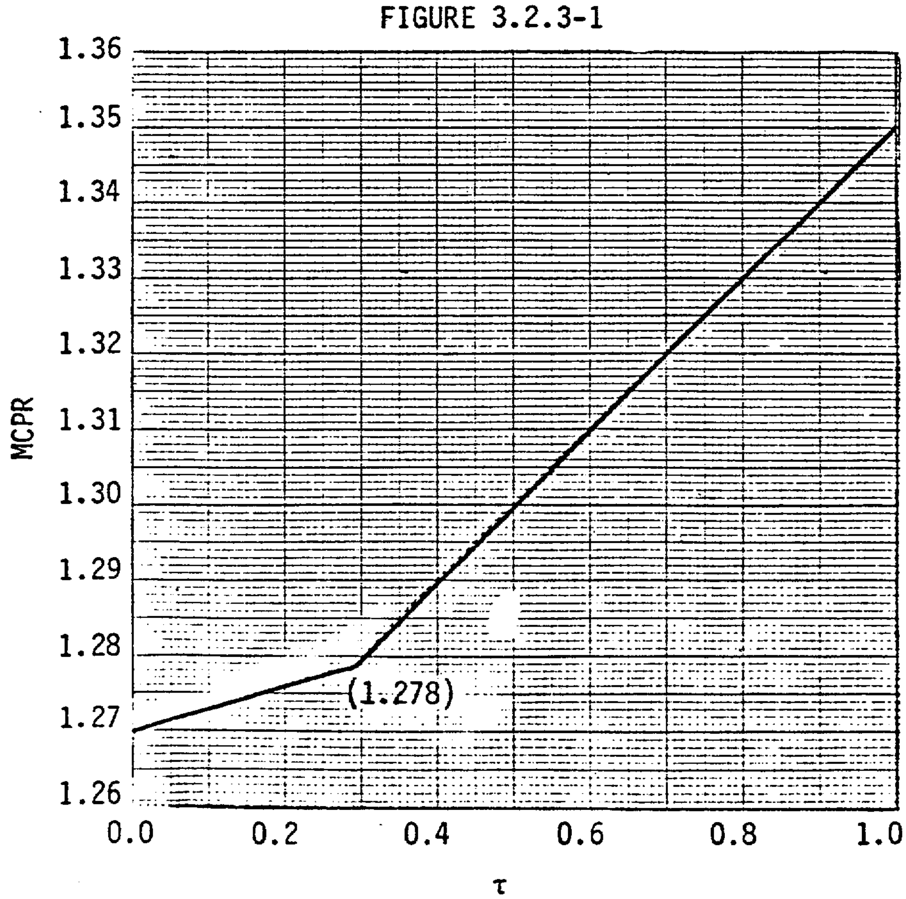
a. At least once per 24 hours,

b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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



MCPR LIMIT FOR 8X8R FUEL AT RATED FLOW
FIGURE 3.2.3-1



MCPR LIMIT FOR P8X8R FUEL AT RATED FLOW
FIGURE 3.2.3-2

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in the Final Acceptance Criteria (FAC) issued in June 1971 considering the postulated effects of fuel pellet densification.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming an LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5, is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses performed with Reference 1 are: (1) the analysis assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5; (2) fission product decay is computed assuming an energy release rate of 200 MEV/fission; (3) pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; and (4) the effects of core spray entrainment and counter-current flow limitation as described in Reference 2, are included in the reflooding calculations.

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

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR HATCH-UNIT 2

Plant Parameters:

Core Thermal Power .... 2531 Mwt which corresponds to 105% of license core power*

Vessel Steam Output .... 10.96×10^6 lbm/h which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure .... 1055 psia

Design Basis Recirculation Line Break Area For:

a. Large Breaks .... 4.0, 2.4, 2.0, 2.1 and 1.0 ft^2

b. Small Breaks .... 1.0, 0.9, 0.4 and 0.07 ft^2

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.18

A more detailed list of input to each model and its source is presented in Section II of Reference 1 and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification linear heat generation rate limit.

TABLE 3.3.5-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

NOTE

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

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

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

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

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

f. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.11.1 or 3.9.11.2.

TABLE 3.3.5-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. APRM		
a. Flow Referenced Simulated Thermal Power - Upscale	≤ (0.66 W + 42%)*	≤ (0.66 W + 42%)*
b. Inoperative	NA	NA
c. Downscale	≥ 3/125 of full scale	≥ 3/125 of full scale
d. Neutron Flux - High, 12%	≤ 12/125 of full scale	≤ 12/125 of full scale
2. ROD BLOCK MONITOR		
a. Upscale	≤ (0.66W + 41%) Not to exceed 107%	≤ (0.66 W + 41%)
b. Inoperative	NA	NA
c. Downscale	≥ 3/125 of full scale	≥ 3/125 of full scale
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1×10^5 cps	≤ 1×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 3 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 of full scale	≤ 108/125 of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 of full scale	≥ 5/125 of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 36.2 gallons	≤ 36.2 gallons

* The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 28 TO FACILITY OPERATING LICENSE NO. NPF-5

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

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2
DOCKET NO. 50-366

1.0 Introduction

By letter dated March 11, 1982 (Ref. 1) Georgia Power Company (the licensee) proposed changes to the Technical Specifications (TSs) of Hatch 2. The proposed changes relate to the core for Cycle 3 operation at power levels up to 2,436 MWt (100% power). In support of the reload application, the licensee enclosed proposed TS changes in Reference 1 and the General Electric (GE) BWR supplemental licensing submittal (Ref. 2).

This reload involves loading of prepressurized GE 8x8 retrofit (P8x8R) fuel. This is the same type of fuel as was loaded during the previous reload. The description of the nuclear and mechanical designs of 8x8 retrofit is contained in References 3 and 4. Reference 3 also contains a complete set of references of topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing a mixture of fuel. The use and safety implications of prepressurized fuel have been found acceptable per Reference 4. The conclusions of Reference 5 found that the methods of Reference 3 were generally applicable to prepressurized fuel. Therefore, unless otherwise specified, Reference 3, as supported by Reference 5, is adequate justification for the current application of prepressurized fuel.

2.0 Evaluation

2.1 Reactor Physics

The reload application follows the procedure described in NEDE-24011-P-A-2, "Generic Reload Fuel Application". We have reviewed this application and the consequent TS changes. The transient analysis input parameters are typical for BWRs and are acceptable. Core wide transient analysis results are given for the limiting transients, and the required operating limit values for minimum critical power ratio (MCPR) are given for each fuel type. The revised MCPR limits are required by the reload, and they are acceptable.

2.2 Thermal Hydraulics

As stated in Reference 3, for BWR cores which reload with GE's retrofit 8x8R fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient or fuel misloading, the most limiting events have been reanalyzed for this reload by the licensee in order to determine which event results in the largest reduction in the MCPR. These events have been analyzed for the exposed fuel and fresh fuel. Addition of the largest reductions in critical power ratio to the SLMCPR was used in the MCPR TS to establish the operating limits for each fuel type.

We have found the methods used for this analysis consistent with previously approved past practice (Ref. 3). Wc have found the results of this analysis and the corresponding TS changes acceptable.

2.3 Rod Block Monitor

The licensee proposed to place an upscale trip setpoint limit of 107% power on the Rod Block Monitor in Table 3.3.5-2 of the TSs. This proposal would prevent reactor operation above 107% power. The fuel thermal limit margins are not reduced by this change, none of the present rod block or reactor trip setpoints are affected, and the 107% power limit provides protection to the fuel for an above rated core flow condition. We conclude, for the reasons stated above, that this change is acceptable as it provides for additional fuel integrity during a core flow condition exceeding rated flow.

3.0 Environmental Considerations

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

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 accidents previously considered and does not involve a significant decrease in a safety margin, 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: May 6, 1982

The following NRC staff personnel have contributed to this Safety Evaluation: Morton B. Fairtile.

REFERENCES

1. Letter, Georgia Power Company to Office of Nuclear Reactor Regulation (USNRC), dated March 11, 1982.
2. "Supplemental Reload Licensing Submittal for Hatch Nuclear Station, Unit 2, Reload 2", General Electric Report Y1003J01A32, dated December 1981.
3. "General Electric Boiling Water Reactor Generic Reload Fuel Application", NEDE-24011-P-A-2 and NEDE-24011-A-2, July 1981.
4. Letter, R. E. Engel (GE) to U. S. Nuclear Regulatory Commission, dated January 30, 1979.
5. Letter, T. A. Ippolito (USNRC) to R. Gridley (GE), April 16, 1979, and enclosed SER.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-366

GEORGIA POWER COMPANY, ET AL.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 28 to Facility Operating License No. NPF-5, issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, which revised Technical Specifications (TSs) for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 2 (the facility) located in Appling County, Georgia. The amendment is effective as of the date of issuance.

This amendment revises the TSs to enable operation of the facility after the Cycle 3 Reload, at licensed power.

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

The Commission has determined that the issuance of this 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 issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 11, 1982, (2) Amendment No. 28 to License No. NPF-5, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 6th day of May 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing