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Docket No. 50-366

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. William Widner
Vice President - Engineering
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Widner:

The Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit No. 2. The amendment consists of changes to the license and the Technical Specifications in response to your application dated October 17, 1980, and supplement dated January 30, 1981.

The amendment changes the Technical Specifications to establish revised safety and operating limits for Hatch Unit No. 2 operation during Cycle 2 with Reload 1 fuel inserted. The amendment also removes three satisfied license conditions as required for operation beyond the first cycle.

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

Sincerely,

Original signed by
Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 21 to NPF-5
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555
February 11, 1981

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Docket No. 50-366

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: HATCH UNIT NO. 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

XX Notice of Issuance of Facility Operating License(s) or Amendment(s).

XX Other: Amendment No. 21
Referenced documents have been provided PDR

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#4:DL					
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DATE→	2/11/81					

NRC FORM 102 (1-76)

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

February 10, 1981

Docket No. 50-366

Mr. William Widner
Vice President - Engineering
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Widner:

The Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit No. 2. The amendment consists of changes to the license and the Technical Specifications in response to your application dated October 17, 1980, and supplement dated January 30, 1981.

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

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 21 to NPF-5
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Hatch 1/2
Georgia Power Company

50-321/366

cc w/enclosure(s):

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

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

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

Director, Criteria and Standards
Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

cc w/enclosure(s) & incoming dtd.:
10/17/80 & 1/30/81

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Room 610
270 Washington Street, S.W.
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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. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
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 October 17, 1980, as supplemented January 30, 1981, 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, Facility Operating License No. NPF-5 is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and as follows:

   A. Revise paragraph 2.C.(2) to read:

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 21 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   B. Delete paragraphs 2.C.(3)(a), 2.C.(3)(c) and 2.C.(3)(d).

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 10, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 21

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

2-1

B2-1

B2-4

B2-9

3/4 2-1

3/4 2-4A (new)

3/4 2-6

3/4 2-7

3/4 2-7a (new)

3/4 2-7b (new)

B3/4 2-1

B3/4 2-3

5-1

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER (Low Pressure or Low Flow)

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

THERMAL POWER (High Pressure and High Flow)

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 AND 2.

ACTION:

With MCPR less than 1.07 and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure $\leq$ 1325 psig with 2 hours.

2.1 SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the low pressure ECCS to restore the reactor vessel water level, after depressurizing the reactor vessel, if required.

BASES

2.0 The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07. MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER (Low Pressure or Low Flow)

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure blow 785 psig is conservative.

BASES (Continued)

2.1.2 THERMAL POWER (High Pressure and High Flow)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[a], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), GEXL correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia
Mass Flow:	0.1 to 1.25 10^6 lb/hr-ft^2
Inlet Subcooling:	0 to 100 Btu/lb
Local Peaking:	1.61 at a corner rod to 1.47 at an interior rod

(a) General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design application," NEDO-10958 and NEDE-10958.

2.1.2 THERMAL POWER (High Pressure and High Flow) (Continued)

	Shape	Max/Avg.
Axial Peaking:	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array	64 Rods in an 8 x 8 array	

The required input to the statistical model are the uncertainties listed in Bases Table B 2.1.2-1, the nominal values of the core parameters listed in Bases Table B 2.1.2-2, and the relative assembly power distribution shown in Bases Table B 2.1.2-3. Bases Table B 2.1.2-4 shows the R-factor distributions that are input to the statistical model which is used to establish the Safety Limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

The bases for the uncertainties in the core parameters are given in NEDO-20340[b], and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958 [a]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Hatch - Unit 2 during any fuel cycle would not be as severe as the distribution used in the analysis. The pressure Safety Limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

(a) "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and design application," NEDO-10958 and NEDE-10958.

(b) General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Admendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.

Bases Table B 2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT*

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	8.7
R Factor	1.6
Critical Power	3.6

*The uncertainty analysis used to establish the core wide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core.

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits. Operation with a trip set less conservative than its Trip Setpoint, but within its specified Allowable Value, is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

1. Intermediate Range Monitor, Neutron Flux

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus, as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed, Section 7.5 of the FSAR. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM's are not yet on scale. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 1% of RATED THERMAL POWER, thus maintaining MCPR above 1.07. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15/125 divisions of full scale neutron flux provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RSCS and RWM.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM flux scram trip in the Run mode consists of a flow referenced simulated thermal power scram setpoint and a fixed neutron flux scram setpoint. The APRM flow referenced neutron flux signal is passed through a filtering network with a time constant which is representative of the fuel dynamics. This provides a flow referenced signal that approximates the average heat flux or thermal power that is developed in the core during transient or steady-state conditions.

The APRM flow referenced simulated thermal power scram trip setting at full recirculation flow is adjustable up to 113.5% of RATED THERMAL POWER. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 100°F feedwater heating event, than would result with the 118% fixed neutron flux scram trip. The lower flow referenced scram setpoint therefore decreases the severity, ΔCPR, of a slow thermal transient and allows lower operating limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the fuel cycle.

The APRM fixed neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow referenced simulated thermal power scram.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint or APRM gain must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when the CMFLPD exceeds the FRTP.

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, or 3.2.1-4.

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3 or 3.2.1-4, 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.

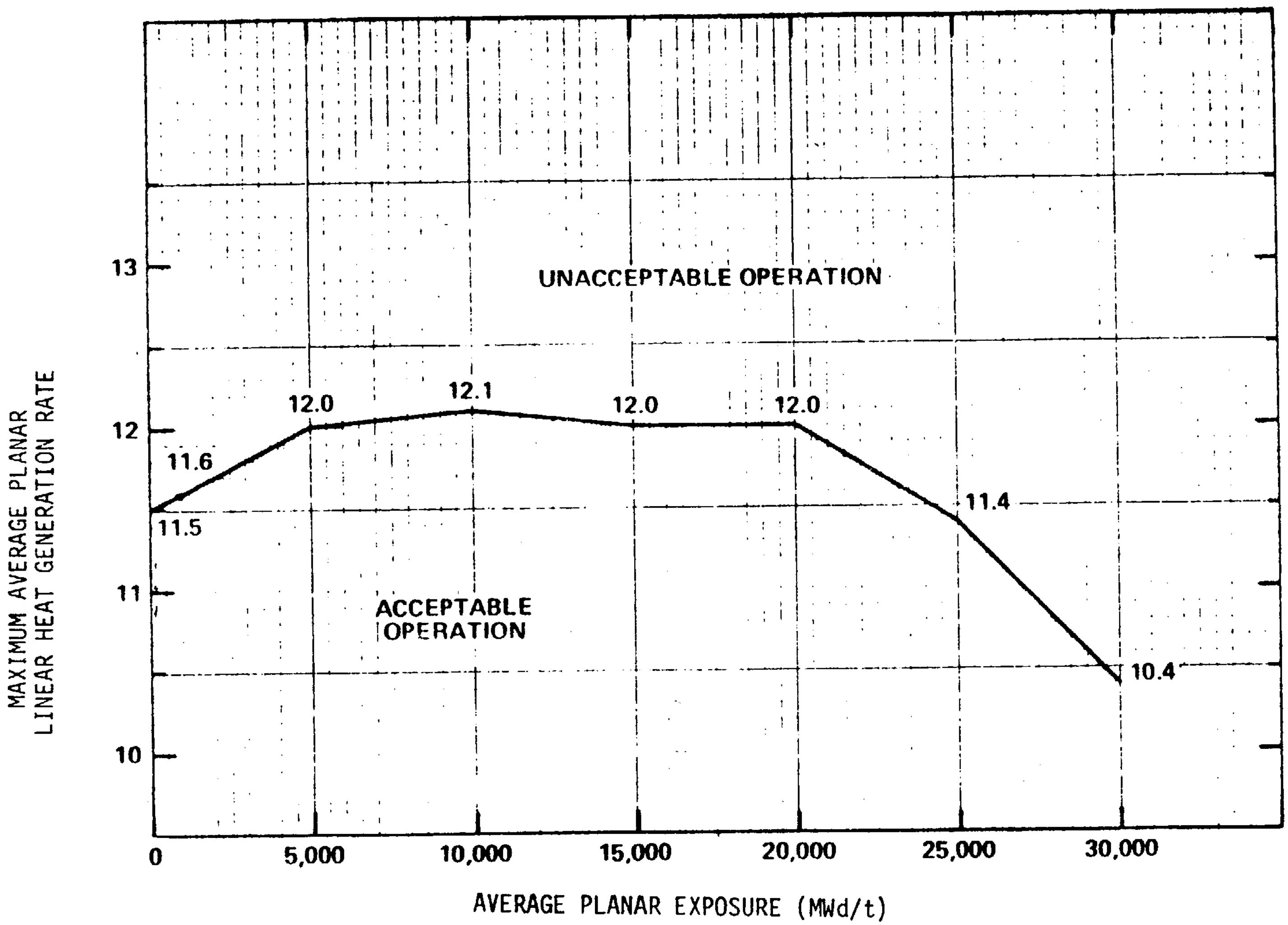
SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

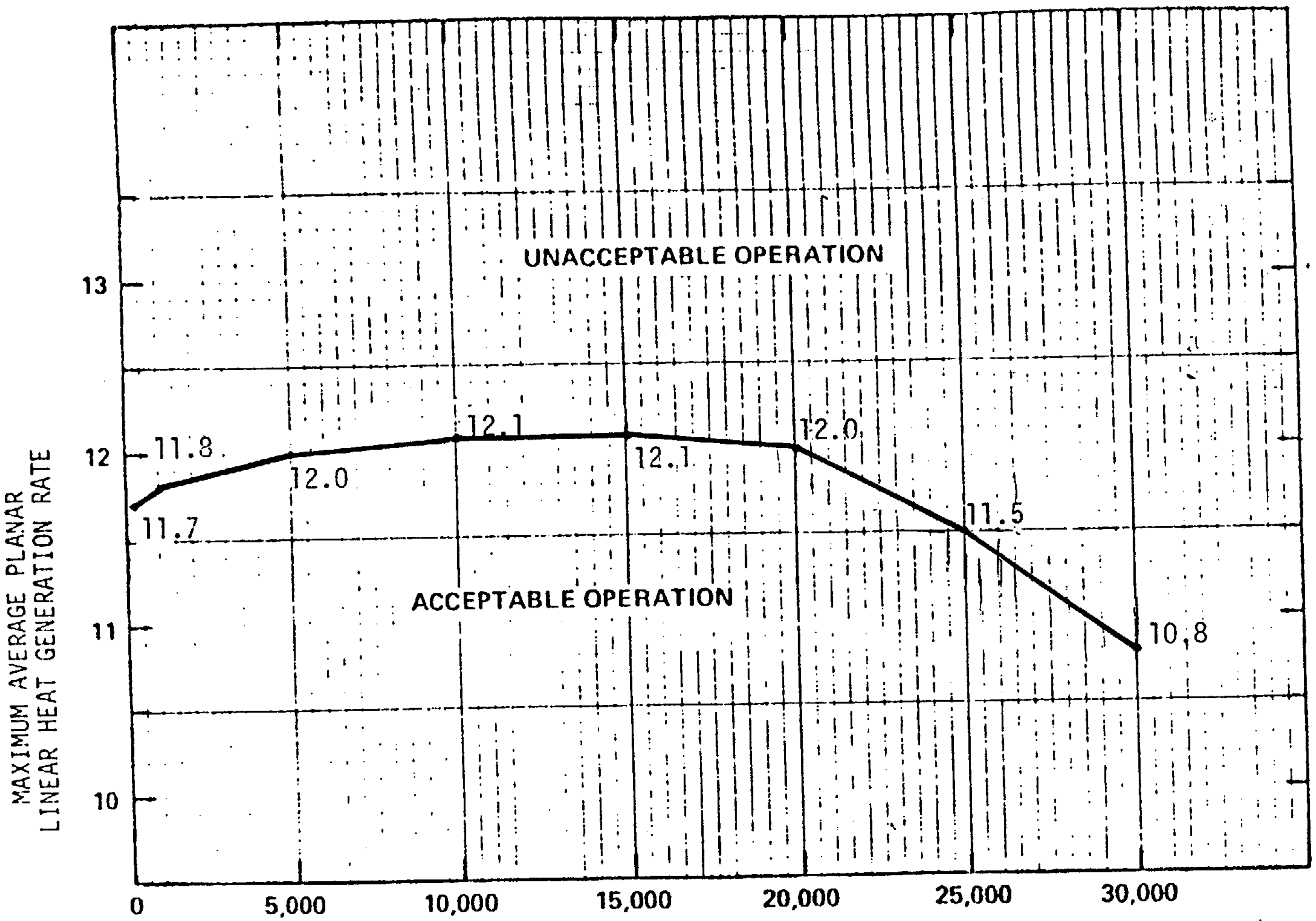
b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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/t)

FUEL TYPE P8DRB284LA

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE
FIGURE 3.2.1-4

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow referenced simulated thermal power scram trip setpoint (S) and control rod block trip setpoint (S_{RB}) shall be established* according to the following relationships:

$$S \leq (0.66W + 51\%)$$

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

where: S and S_{RB} are in percent of RATED THERMAL POWER, and
W = Loop recirculation flow in percent of rated flow.

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

ACTION:

With S or S_{RB} exceeding the allowable value, initiate corrective action within 15 minutes and continue corrective action so that S and S_{RB} are within the required limits* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The CMFLPD shall be determined and the APRM flow referenced simulated thermal power scram and control rod block trip setpoints or APRM readings adjusted, as required:

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 CMFLPD $\geq$ FRTP.

*With CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) greater than the fraction of RATED THERMAL POWER (FRTP), $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, up to 95% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times CMFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of average scram time and core flow, shall be equal to or greater than shown in Figure 3.2.3-1 or Figure 3.2.3-2 multiplied by the K_f shown in Figure 3.2.3-3, where:

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

$$\tau_A = 1.096 \text{ sec (Specification 3.1.3.3 scram time limit to notch 36),}$$

$$\tau_B = 0.834 + 1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{\frac{1}{2}}(0.059),$$

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

n = number of surveillance tests performed to date in cycle,
N_i = number of active control rods measured in the i^{th} surveillance test,
τ_i = average scram time to notch 36 of all rods measured in the i^{th} surveillance test, and
N_1 = total number of active rods measured in 4.1.3.2.a.

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

ACTION:

With MCPR less than the applicable limit determined from Figure 3.2.3-1 or Figure 3.2.3-2 initiate corrective action within 15 minutes and continue corrective action so that MCPR is equal to or greater than the applicable limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 The MCPR limit at rated flow shall be determined for each type of fuel (8X8R and P8X8R) from Figures 3.2.3-1 and 3.2.3-2 using:

a. τ = 1.0 prior to the initial scram time measurements for the cycle performed in accordance with Specification 4.1.3.2.a, or

SURVEILLANCE REQUIREMENTS (Continued)

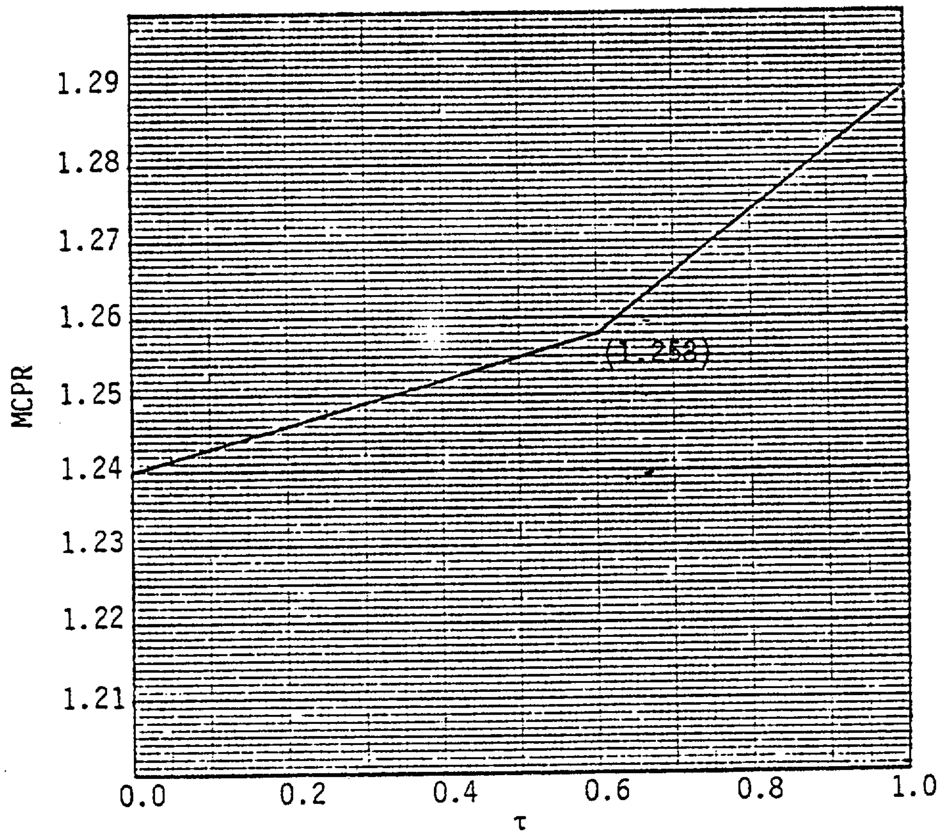
b. τ as defined in Specification 3.2.3; the determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by Specification

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

a. At least once per 24 hours,

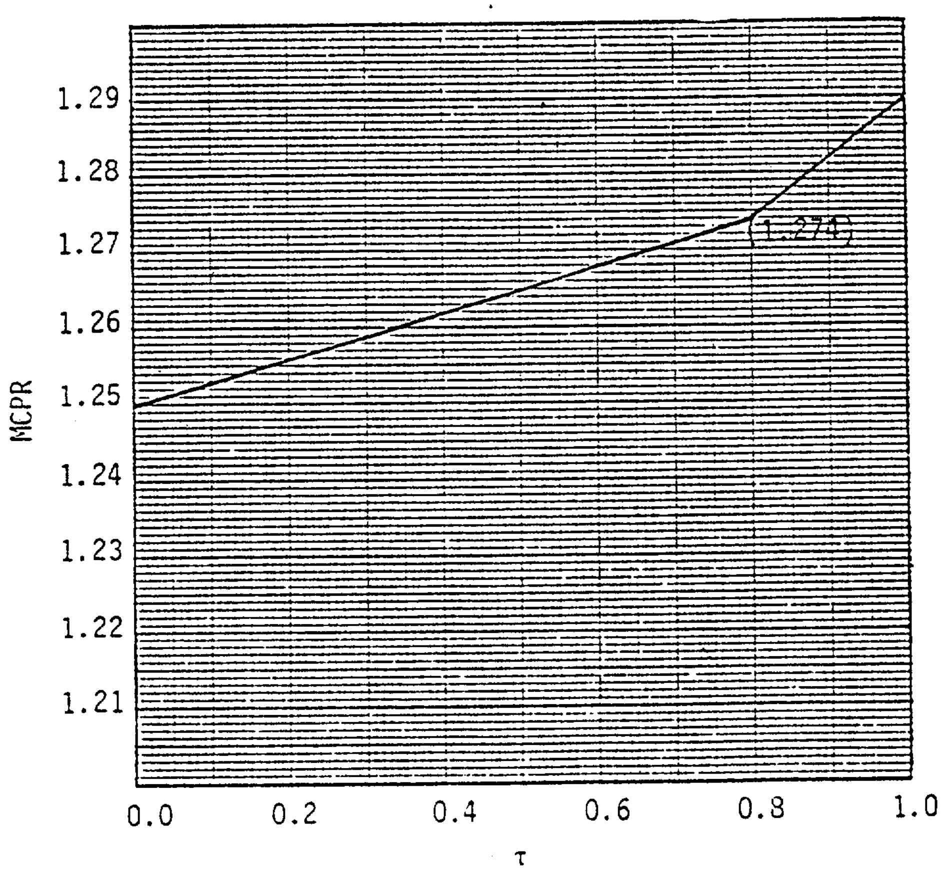
b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

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



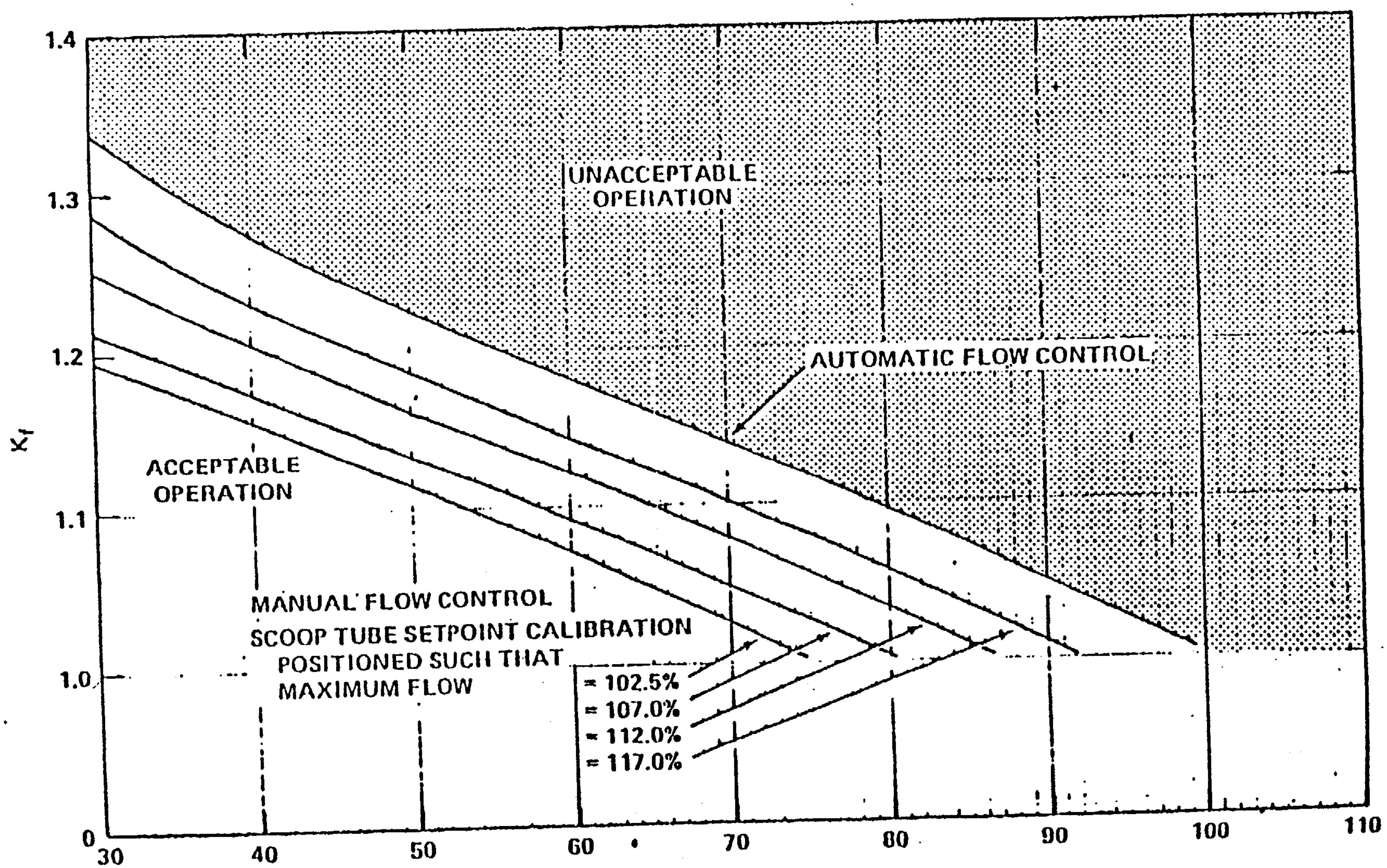
MCPR LIMIT FOR 8X8R FUEL AT RATED FLOW

FIGURE 3.2.3-1



MCPR LIMIT FOR P8X8R FUEL AT RATED FLOW

FIGURE 3.2.3-2



Core Flow, % of Rated Core Flow

K_F Factor

FIGURE 3.2.3-3

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 All LINEAR HEAT GENERATION RATES (LHGRs) shall not exceed 13.4 Kw/ft.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and continue corrective action so that the LHGR is within the limit within 2 hours or reduce THERMAL POWER to less then 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit;

a. At least once per 24 hours,

b. When 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 on a LIMITING CONTROL ROD PATTERN for LHGR.

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 and 3.2.1-4.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3 and 3.2.1-4 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 and 3.2.1-4; (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.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The scram setting and rod block functions of the APRM instruments or APRM readings 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 or APRM readings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CMFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

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. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting as given in Specification 2.2.1.

To assure that the fuel cladding integrity Safety Limits are not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which results 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 load rejection trip with failure of the turbine bypass. This transient yields the largest Δ CPR. When added to the Safety Limit MCPR of 1.07 the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

MINIMUM CRITICAL POWER RATIO (Continued)

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.1-6 that are input to a GE-core dynamic behavior transient computer program described in NEDO-10802[3]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDO-20566[1]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

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

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

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

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

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone coincides with the exclusion area and is also shown in Figure 5.1.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel structure composed of a series of vertical right cylinders and truncated cones which form a drywell. This drywell is attached to a suppression chamber through a series of vents. The suppression chamber is a steel pressure vessel in the shape of a torus. The primary containment has a total minimum free air volume of 255,978 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum design internal pressure 56 psig.

b. Maximum allowable internal pressure 62 psig.

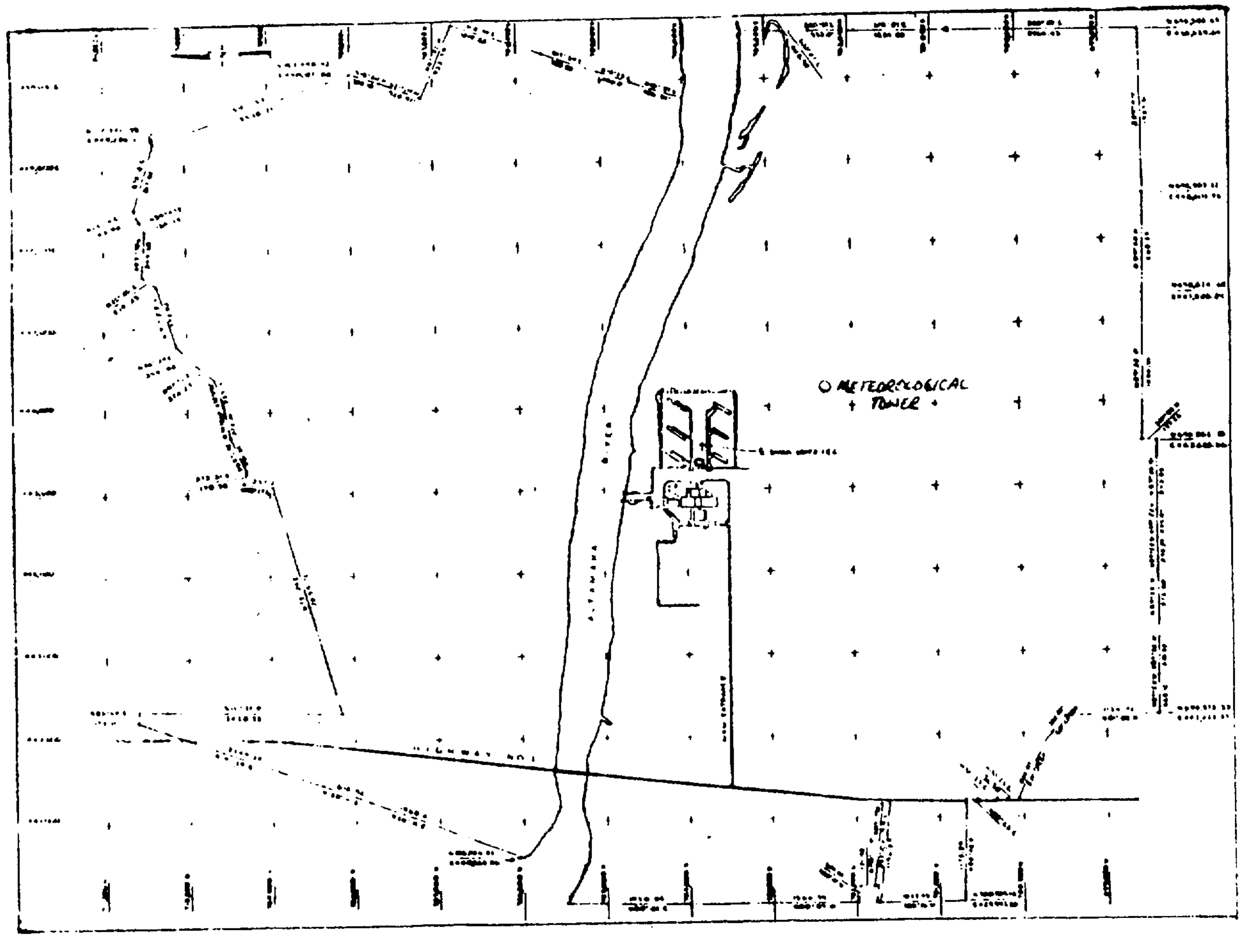
c. Maximum internal temperature 340°F.

d. Maximum external pressure 2 psig.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 560 fuel assemblies with each fuel assembly containing 62 fuel rods and 2 water rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight percent of 3341 grams uranium. The initial core loading shall have a maximum average enrichment of 1.87 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum average enrichment of 2.90 weight percent U-235.



EXCLUSION AREA AND LOW POPULATION ZONE

FIGURE 5.1.1-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 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. INTRODUCTION

By letter dated October 17, 1980, Georgia Power Company (the licensee) requested revisions to the Technical Specifications (TSs) appended to Facility Operating License No. NPF-5 to complete the first refueling of the Edwin I. Hatch Nuclear Plant, Unit No. 2 (Hatch 2), and begin Cycle 2 operation (Ref. 1). The original submittal was revised on January 30, 1981 (Ref. 2) to take advantage of the fact that actual control blade scram times to 20% insertion are typically faster than the scram time assumed in the licensing analyses, and the minimum critical power ratio (MCPR) operating limits calculated using actual scram time data are less limiting than those derived using the NRC staff's conservative penalty factors. The proposed amendment is supported by General Electric Company's (GE) plant-specific reload report (Ref. 3).

In addition to the routine considerations in any reload application, the submittal also addressed the NPF-5 license conditions: 2.C.(3)(a) Fuel Performance, 2.C.(3)(c) Abnormal Operational Transient Reanalysis, and 2.C.(3)(d) Boiling Transition Data, which are eligible for deletion.

The Hatch 2 Reload 1 involves loading 164 (P8x8R) fuel bundles of type P8DRB284LA. The remainder of the 560 fuel bundles in the core will be fuel used during Cycle 1.

2. EVALUATION

2.1 TRANSIENTS

Various transient events will reduce the MCPR from its operating value. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational occurrence, the most limiting transients (rod withdrawal error, loss of feedwater heating and the pressurization events) have been reanalyzed by the licensee. This reanalysis calculates the reduction

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in critical power ratio (CPR) for each of the most limiting transients and the largest is used to establish the operating limit MCPR. Each of these events has been conservatively analyzed for each of the fuel types, i.e., 8x8R and P8x8R, and for the full range of exposure through the cycle. The analysis shows that the most limiting transients for this cycle are the pressurization events, load rejection and feedwater controller failure.

For the analysis of the limiting pressurization events, the licensee used ODYN per the requirements of our acceptance of this code (Ref. 4 and 5). The licensee has proposed a change to the MCPR TS. The change assures that the requirements for credit of scram speed are satisfied and that the MCPR values are acceptable over the range of cycle conditions and fuel types.

2.2 ACCIDENT ANALYSIS

2.2.1 EMERGENCY CORE COOLING SYSTEM (ECCS) PERFORMANCE ANALYSIS

The licensee has reevaluated ECCS performance for the new reload fuel design by methods that have been previously accepted by the NRC staff. The results of the plant-specific analysis are given in Section 14 of Ref. 3. We have reviewed the information that has been submitted by the licensee and have concluded that all requirements of 10 CFR 50.46 and its Appendix K will be met when the reactor is operated in accordance with the proposed maximum average planar linear heat generation rate (MAPLHGR) limits versus average planar exposure values. These MAPLHGR limits vs. average planar exposure curves have been incorporated in the revised TSs.

2.2.2 CONTROL ROD DROP ACCIDENT

Because the characteristic accident analysis input parameters for the worst case control rod drop accident were not bounded by all the assumptions of the bounding analysis, the licensee reanalyzed this accident on a plant-specific basis. The results showed the peak fuel enthalphy to be less than the 280 calories per gram limit.

2.2.3 FUEL LOADING ERROR

The licensee has considered the effect of a possible fuel loading error on bundle CPR. An analysis of the most severe misoriented and mislocated fuel loading error per the accepted version of the generic reload topical (Ref. 6), shows that the worst possible rotation or mislocation of a fuel bundle will not cause a violation of the safety limit MCPR.

2.2.4 OVERPRESSURIZATION ANALYSIS

The overpressurization analysis of the main steamline isolation valve closure with high flux scram, which is the limiting overpressure event, has been performed with the ODYN code. The analysis shows an acceptable margin to the overpressurization limit that is adequate to account for the failure of one safety valve.

2.2.5 THERMAL-HYDRAULIC STABILITY

A thermal-hydraulic stability analysis was performed with the methods described in Ref. 6. The results show that the channel hydrodynamic and reactor core decay ratios at the least stable operating state are below the ultimate performance limit decay ratio of 1.0. (The least stable operating state corresponds to the intersection of the natural circulation curve and the 105% rod line on the power-flow map.)

Generic concerns on operation at natural circulation conditions have been raised due to increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. These concerns relate to both the consequences of operation at decay ratios of 1.0 and the capability of the analytical methods to accurately predict decay ratios.

A requirement to preclude normal operation in the natural circulation mode has been instituted in the plant TSs. This restriction continues to provide a significant increase in reactor stability operating margins and is acceptable.

2.2.6 CONCLUSION

Based on the foregoing, the proposed TSs and supporting analysis are acceptable.

2.3 LICENSE CONDITION 2.C.(3)(a), "FUEL PERFORMANCE"

Condition 2.C.(3)(a) of the Hatch 2 license states:

> "Georgia Power Company shall, prior to startup for that cycle of operation in which burnups greater than 20,000 megawatt days per ton of uranium are expected to be attained, provide for Commission review and obtain Commission approval of GEGAP-III calculations and other affected analyses utilizing fission gas release calculational methodology approved for burnups greater than 20,000 megawatt days per ton of uranium."

This license condition was imposed as a result of our concern (Ref. 7) that fission gas release from the fuel may not be correctly calculated for burnups above 20,000 MWd/MTu.

The licensee has elected (Ref. 1) not to provide revised calculations which account for enhanced fission gas release at high burnups for the proposed cycle of operation. As a basis for this decision, the licensee cited an NRC letter dated March 10, 1980 (Ref. 8) on this subject. The NRC letter, in turn, cites a GE letter (Ref. 9) that describes calculations to 33,000 MWd/MTu for a number of GE plant types and fuel designs. These calculations have been accepted (Ref. 8) on an interim basis for continued operation at other Boiling Water Reactor (BWR) facilities and would also apply to Hatch 2 for Cycle 2 operation.

The licensee has also noted (Ref. 1) that a revised GE fuel performance model, GESTR, is currently under NRC staff review. Until such time that the GESTR model is approved and incorporated into plant safety analyses, it is our intent to require revised calculations that account for burnup effects on fission gas release only when anticipated conditions approach or exceed 33,000 MWd/MTu, the highest value considered for Loss of Coolant Accident (LOCA) analysis in Ref. 4. Hatch 2 will not approach this value in Cycle 2. Since the present MAPLHGR terminates at 30,000 MWd/MTu, no other limit is needed. We therefore conclude that Condition 2.C.(3)(a) may be deleted from the Hatch 2 operating license.

2.4 LICENSE CONDITION 2.C.(3)(c), "ABNORMAL OPERATIONAL TRANSIENT REANALYSIS"

Facility Operating License No. NPF-5 Condition 2.C.(3)(c) requires a reanalysis of the limiting abnormal operational transient using the GE One Dimensional Core Transient Model ODYN computer code prior to startup following the first refueling outage for Hatch 2.

We reviewed results of the abnormal transient reanalysis using the ODYN computer code (Ref. 3), which show that the operating MCPR limit of 1.24 for 8x8R/P8x8R fuel from beginning of cycle (BOC) to end of cycle (EOC) is acceptable (Ref. 10). We note that the application of additional conservatism using adjustment factors based on measured scram time data resulted in a MCPR value for P8x8R fuel of 1.25. This conservative result appears in TS Figure 3.2.3-2 and was caused by the statistical treatment of round-off error. We find that the operating MCPR limits computed by the ODYN code are acceptable and have been applied conservatively. We therefore conclude that Condition 2.C.(3)(c) may be deleted from the Hatch 2 operating license.

2.5 LICENSE CONDITION 2.C.(3)(d), "BOILING TRANSITION DATA"

Condition 2.C.(3)(d) of the Hatch 2 license states:

> "Georgia Power Company shall, prior to startup following the first refueling outage, provide for Commission review and obtain Commission approval for the use of boiling transition data for 8x8 fuel bundles with two water rods in order to support the use of the GEXL correlation for fuel bundle radial peaking patterns expected to be encountered during operation beyond the first cycle."

The concern with boiling transition data for 8x8R fuel and the GEXL correlation was documented during the original Hatch 2 licensing process (Ref. 11 and 12).

We have reviewed the GE submittals on this subject (Ref. 13, 14 and 15) and found that the GEXL correlation for fuel bundle radial peaking factors is acceptable for 8x8R fuel reload application (Ref. 16).

As stated in Ref. 11 for BWR cores which reload with GE's retrofit P8x8R fuel, the allowable MCPR, resulting from either core-wide or localized abnormal operational transients, is equal to 1.07. With this MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) proposed by the licensee for Cycle 2 represents a .01 increase from the 1.06 SLMCPR applicable during Cycle 1. The basis for the revised safety limit is addressed in Ref. 17, while our generic approval of the new limit is given in Ref. 11. This change is consistent with the criteria of Standard Review Plan 4.4 and on that basis has been found acceptable in Ref. 11.

We therefore conclude that Condition 2.C.(3)(d) may be deleted from the Hatch 2 operating license.

ENVIRONMENTAL CONSIDERATIONS

We have determined that this 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 this amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 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.

CONCLUSIONS

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.

REFERENCES

1. W. A. Widner (GPC) letter to the U. S. N. R. C. dated October 17, 1980.

2. W. A. Widner (GPC) letter to the U. S. N. R. C. dated January 30, 1981.

3. "Supplemental Reload Licensing Submittal for Hatch Nuclear Power Station Unit 2 Reload-1," Y1003J01A10, July 1980.

4. Memorandum from P. S. Check to T. M. Novak and R. L. Tedesco, "Safety Evaluation for 'Qualification of the One-Dimensional Core Transient Model For Boiling Water Reactors,' NEDO-24154 and NEDE-24154P, Volume I, II and III," October 22, 1980.

5. Memorandum from P. S. Check to T. M. Novak and R. L. Tedesco, "Supplemental Safety Evaluation for ODYN Code," November 20, 1980.

6. General Electric Company Topical Report NEDE-24011-P-A, "Generic Reload Fuel Application."

REFERENCES (Cont'd.)

7. D. F. Ross, Jr. (NRC) memorandum to D. B. Vassallo (NRC) on "SER Input for Hatch, Unit 2" dated April 25, 1977.

8. T. A. Ippolito (NRC) letter to C. F. Whitner (GPC) dated March 10, 1980.

9. G. G. Sherwood (GE) letter to D. F. Ross, Jr. (NRC) dated December 22, 1976.

10. Memorandum for T. Novak from P. S. Check, SSER for ODYN Code, November 20, 1980.

11. Letter, D. Eisenhut (NRC) to R. Gridley (GE), "Safety Evaluation for the General Electric Topical Report, Generic Reload Fuel Application NEDE-24011-P," May 12, 1978.

12. Safety Evaluation Report Related to Operation of Edwin I. Hatch Nuclear Plant Unit No. 2, NUREG-0411, June 1978.

13. Letter, R. E. Engel (GE) to L. S. Rubenstein (NRC), Response to NRC Concerns on the 8x8R GEXL Correlation, August 26, 1980.

14. Letter, J. F. Quirk (GE) to O. D. Parr (NRC), "General Electric Licensing Topical Report, NEDE-24011-P-A, Generic Reload Fuel Application, Appendix D," February 28, 1979.

15. Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "General Electric Licensing Topical Report NEDE-24011-P-A, Generic Reload Fuel Application Appendix D Submittal," December 14, 1979.

16. Memorandum for T. Novak from L. Rubenstein, SER for the GEXL Correlation for 8x8R Fuel Reload Application per the Appendix D submittals of Generic Reload Fuel Application NEDE-24011-P dated February 28, 1979 and December 14, 1979, February 1981.

17. "Generic Reload Fuel Application," General Electric Report, NEDE-24011-P-3, March 1978.

Dated: February 10, 1981

7590-01

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.21 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 the license and the Technical Specifications 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 its date of issuance.

The amendment revises the Technical Specifications to establish revised safety and operating limits for Hatch Unit No. 2 operation during Cycle 2 with Reload 1 fuel inserted. The amendment also removes three satisfied license conditions as required for operation beyond the first cycle.

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.

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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated October 17, 1980, as supplemented January 30, 1981, (2) Amendment No. 21 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 10th day of February 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing