6/1/76

Docket No. 50-321

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Georgia Power Company
Oglethorpe Electric Membership Corporation
ATTN: Mr. I. S. Mitchell, III
Vice President and Secretary
Georgia Power Company
Atlanta, Georgia 30302

Gentlemen:

The Commission has issued the enclosed Amendment No. 33 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated April 9, 1976.

This amendment will incorporate revised operating limits into the Technical Specifications based upon an Emergency Core Cooling System (ECCS) Analysis which takes credit for the improved post-loss of cooling accident (LOCA) core reflood capability resulting from recently completed modifications to the Low Pressure Coolant Injection (LPCI) system.

In reviewing your application it was found that certain changes in the proposed Technical Specifications were required. These changes were discussed with and approved by your staff.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 33
2. Safety Evaluation
3. Federal Register Notice

assuming input from ECCS people

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GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company and Oglethorpe Electric Membership Corporation (the licensees) dated April 9, 1976, 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; and

   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.

   E. After weighing the environmental aspects involved, 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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 33

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace pages 1.1-6, 1.1-7, 1.1-8, 3.5-3, 3.5-4, 3.5-15, 3.6-22, 3.11-3, 3.11-4, 3.11-5, Figure 3.11-1 (Sheet 1) and Figure 3.11-1 (Sheet 2) with the attached revised pages. Add page 3.11-6.

1.1 FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure >800 psia and Core Flow >10% of Rated

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 critical power ratio 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 (1), 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 flux:	0.1 to 1.25 10^6lb/hr	
Inlet Subcooling:	0 to 100 Btu/lb	
Local Peaking:	1.61 at a corner rod to 1.47 at an interior rod	
Axial Peaking:	Shape	Max/Avg.
	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 8x8 array 49 Rods in a 7x7 array	

The required inputs to the statistical model are the uncertainties listed in Table 5-1 of reference 2 and the nominal values of the core parameters listed in Table 5-2 of reference 2.

1.1.A. Reactor Pressure $>$ 800 psia and Core Flow $>$ 10% of Rated (Cont'd)

The basis for the uncertainties in the core parameters is given in NEDO-20340[3] and the basis for the uncertainity in the GEXL correlation is given in NEDO-10958[1]. 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 No. 1 during any fuel cycle would not be as severe as the distribution used in the analysis.

B. Core Thermal Power Limit (Reactor Pressure $<$ 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 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 core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

D. Reactor Water Level (Hot or Cold Shutdown Condition)

For the fuel in the core during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety limit has been established at 25 inches above the top of the irradiated fuel to provide a point which can be monitored and also provide adequate margin.

E. References

1. "General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application", NEDO 10958 and NEDE 10958.

2. "Edwin I. Hatch Nuclear Plant Unit 1 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged", NEDO-21124, November 1975.

3. General Electric "Process Computer Performance Evaluation Accuracy", NEDO-20340, and Amendment 1, NEDO-20340-1, dated June 1974 and December 1974, respectively.

3.5.B 1. Normal System Availability (Continued)

b. One RHR loop with two pumps or two loops with one pump per loop shall be operable in the shutdown cooling mode when irradiated fuel is in the reactor vessel and the reactor pressure is atmospheric except prior to a reactor startup as stated in Specification 3.5.B.1.a.

c. The reactor shall not be started up with the RHR system supplying cooling to the fuel pool.

d. During reactor power operation, the LPCI system discharge cross-tie valve, E11-F010, shall be in the closed position and the associated valve motor starter circuit breaker shall be locked in the off position. In addition, an annunciator which indicates that the cross-tie valve is not in the fully closed position shall be available in the control room.

2. Operation with Inoperable Components

a. One LPCI Pump Inoperable

If one LPCI pump is inoperable, the reactor may remain in operation for a period not to exceed seven (7) days provided that the remaining LPCI pumps, both LPCI subsystem flow paths, the Core Spray system, and the associated diesel generators are operable.

b. One LPCI Subsystem Inoperable

A LPCI sybsystem is considered to to inoperable if (1) both of the LPCI pumps within that system are inoperable or (2) the active valves in the subsystem flow path are inoperable.

If one LPCI subsystem is inoperable, the reactor may remain in operation for a period not to exceed seven (7) days provided that all active components of the remaining LPCI subsystem, the Core Spray system, and the associated diesel generators are operable.

4.5.B.1. Normal Operational Tests

	Item	Frequency
b.	Simulated Automatic Actuation Test	Once/Operating Cycle
c.	System flow rate: Each RHR pump shall deliver at least 7700 gpm against a system head of at least 20 psig.	Once/3 months
d.	Pump Operability	Once/month
e.	Motor Operated valve operability	Once/month

2. Surveillance with Inoperable Components

a. One LPCI Pump Inoperable

When one LPCI pump is inoperable, the remaining LPCI pumps and associated flow paths, the Core Spray system, and the associated diesel generators shall be demonstrated to be operable immediately and daily thereafter, until the inoperable LPCI pump is restored to normal service.

b. One LPCI Subsystem Inoperable

When one LPCI subsystem is inoperable, all active components of the remaining LPCI subsystem, the Core Spray system, and the associated diesel generators shall be demonstrated to be operable immediately and daily thereafter, until the inoperable LPCI subsystem is restored to normal service.

3.5.B.2. Operation with Inoperable Components (Continued)

DELETED

4.5.B.2. Surveillance with Inoperable Components (Continued)

DELETED

Observation of the stated requirements for the containment cooling mode assures that the suppression pool and the drywell will be sufficiently cooled, following a loss-of-coolant accident, to prevent primary containment overpressurization. The containment cooling function of the RHR system is permitted only after the core has reflooded to the two-thirds core height level. This prevents inadvertently diverting water needed for core flooding to the less urgent task of containment cooling. The two-thirds core height level interlock may be manually bypassed by a keylock switch.

The intent of the RHR system specifications is to prevent operation above atmospheric pressure without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgment based on experiences and supported by availability analysis. Assurance of the availability of the remaining systems is increased by demonstrating operability immediately and by requiring selected testing during the outage period.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core.

2. Operation with Inoperable Components

With one LPCI pump inoperable or one LPCI subsystem inoperable, adequate core flooding is assured by the demonstrated operability of the redundant LPCI pumps and LPCI subsystem, the Core Spray system, and the associated diesel generators. The reduced redundancy justifies the specified 7 day out-of-service period.

3.6.I Jet Pumps (Continued)

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzel riser system failure.

3.6.J. Recirculation Pump Speeds

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out-of-service. Therefore, continuous reactor operation under such conditions should not be permitted until the necessary analyses have been performed, evaluated and determined acceptable. The reactor may, however, operate for periods up to 24 hours with one recirculation loop out-of-service. This short time period permits corrective action to be taken and minimizes unnecessary shutdowns which is consistent with other Technical Specifications. During this period of time the reactor will be operated within the restrictions of the thermal analysis and will be protected from fuel damage resulting from anticipated transients.

3.11 . FUEL RODS

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K, even considering the postulated effects of fuel pellet densification.

The peak cladding temperature 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 only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50, Appendix K limit. The limiting value for APLHGR is shown in Figures 3.11-1, sheets 1 and 2.

The calculational procedure used to establish the APLHGR shown in Figures 3.11-1, sheets 1 and 2, 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 as compared to previous analyses performed with Reference 1 are: (1) The analyses assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figure 3.11.1; (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assembed after nucleate boilding is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting 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 Table 1 of NEDO-21187[3].

3.11.B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 4 and References 5 and 6, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

The required operating limit MCPR as specified in Specification 3.11.C is derived from the established fuel cladding ingetrity Safety Limit MCPR of 1.06 and an analysis of abnormal operational transients presented in Reference 7.

Various transient events will reduce the MCPR below the operating MCPR. To assure that that the fuel cladding integrity safety limit (MCPR of 1.06) is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (Δ MCPR). Addition of the largest Δ MCPR to the safety limit MCPR gives the minimum operating limit MCPR to avoid violation of the safety limit, should the most limiting transient occur. The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The evaluation of a given transient begins with the system initial parameters shown in Table 7-1 of Reference 7 that are input to a GE core dynamic behavior transient computer program described in Reference 8. 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 Reference 1. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The most limiting transient was the turbine trip without bypass transient which results in a ΔMCPR of 0.26. Consequently, the minimum required operating limit MCPR is 1.32.

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% 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 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 for the same postulated transient event.

The K_f factor curves shown in Figure 3.11-3 were developed generically which 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 at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core 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 MCPR's 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.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.11-3, are conservative for Hatch Unit No. 1 operation because the operating limit MCPR of 1.32 is greater than the orginal 1.20 operating limit MCPR used for the generic derivation of K_f.

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at the 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluations below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

D. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e. there is no allowable time in which the plant can knowingly exceed the limiting values for APLHGR, LHGR, and MCPR. It is a requirement, as stated in Specifications 3.11.A, B, and C that if at any time during steady state power operation, it is determined that the limiting values for APLHGR, LHGR, or MCPR are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving operation beyond a specified limit shall be reported as a Reportable Occurrence. If the specified corrective action described in the LCO's was taken, a thirty-day written report is acceptable.

3.11.E References

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566-P, November 1975.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

3. Edwin I. Hatch Nuclear Plant Unit 1 Emergency Core Cooling System Analysis - Appendix K Requirement With Modified Low Pressure Coolant Injection System NEDO-21187, Supplement 1, April 1976.

4. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August, 1973.

5. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 16, 1974 (USA Regulatory Staff).

6. Communication: V.A. Moore to I.S. Mitchell "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

7. "Edwin I. Hatch Nuclear Plant Unit 1 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged," NEDO-21124, November 1975.

8. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

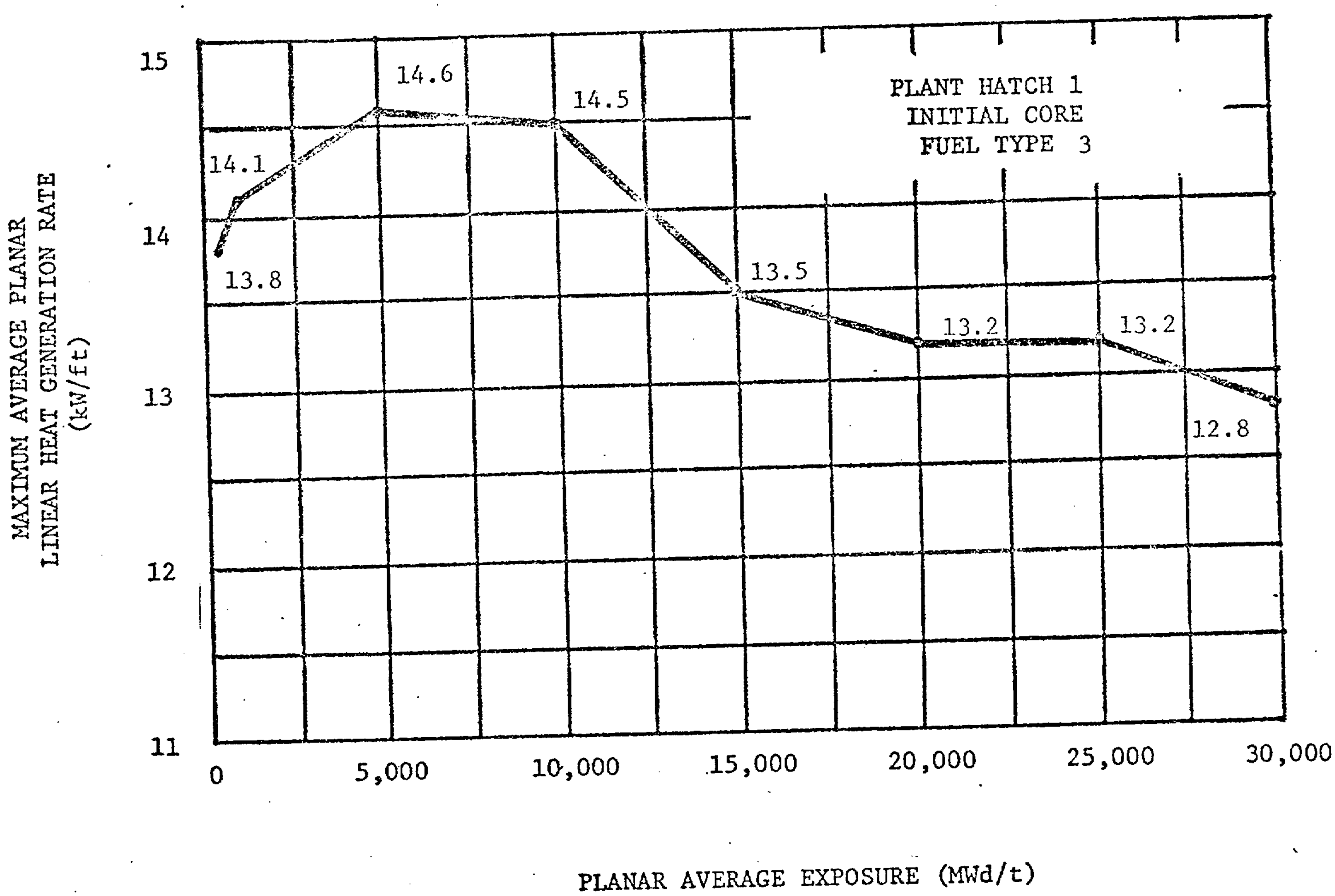

FIGURE 3.11-1 (Sheet 1)

LIMITING VALUE FOR APLHGR

(FUEL TYPE 3)

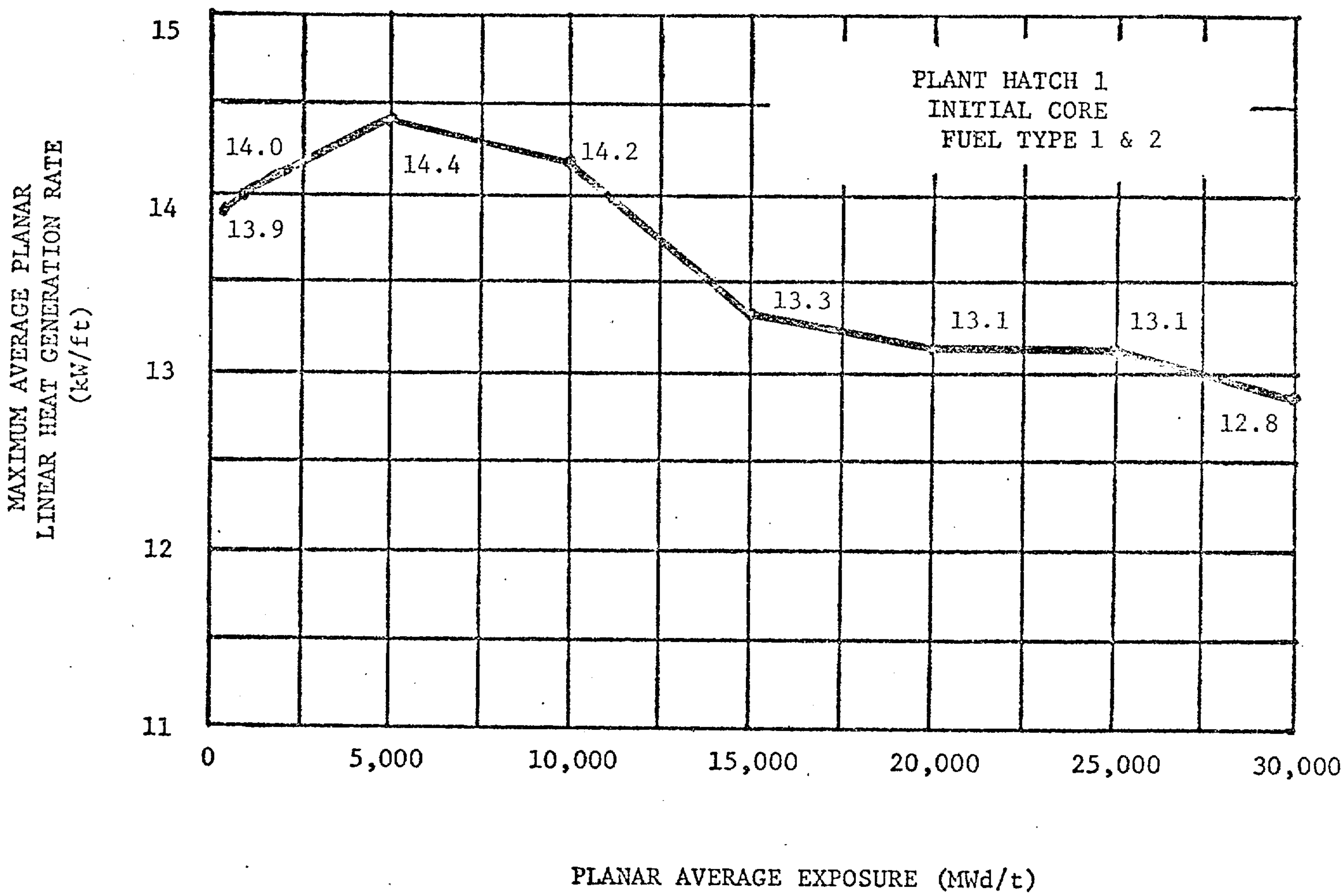

FIGURE 3.11-1 (Sheet 2)

LIMITING VALUE FOR APLHGR

(FUEL TYPE 1 & 2)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 33 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY AND OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-321

Introduction

By letter dated April 9, 1976, Georgia Power Company (GPC) requested an amendment to Facility Operating License No. DPR-57 for Edwin I. Hatch Nuclear Plant Unit No. 1. The proposed license amendment would incorporate revised operating limits into the Hatch Unit No. 1 Technical Specifications based upon an Emergency Core Cooling System (ECCS) Analysis which takes credit for the improved post-loss-of-coolant accident (LOCA) core reflood capability resulting from recently completed modifications to the Low Pressure Coolant Injection (LPCI) system.

Background

On December 17, 1975 the NRC issued License Amendment No. 27 for Hatch Unit No. 1 which authorized operation: (1) with the lower core support bypass flow holes plugged, (2) with operating limits based on the General Electric Thermal Analysis Basis (GETAB), and (3) with operating limits based on an acceptable evaluation model that conformed with the requirements of Section 50.46 and Appendix K of 10 CFR Part 50.

On March 30, 1976, the NRC issued License Amendment No. 31 for Hatch Unit No. 1 which authorized: (1) modifications to the LPCI system which would increase the reliability and capability of the LPCI system to perform its design function in the event of a postulated LOCA, and (2) continued reactor operation, following completion of the LPCI system modifications, in accordance with the ECCS-related operating limits previously established by License Amendment No. 27.

Discussion and Evaluation

The recently completed modifications to the LPCI system involved: (1) elimination of the recirculation loop selection logic, (2) rewiring the system so that the automatic initiation signals direct both of the LPCI system injection valves to open upon detection of LOCA conditions, (3) changing the action of the recirculation loop discharge valves and discharge bypass valves such that they are directed to close upon detection of LOCA conditions, and (4) closing the cross-tie valve between the two LPCI system discharge piping headers and locking open the associated valve motor circuit breaker. These modifications were designed to increase the reliability and availability of the LPCI system in the event of a postulated LOCA, thereby improving the overall performance of the integrated ECCS.

The revised ECCS analysis for Hatch Unit No. 1 with the LPCI modifications referenced another BWR-4 (Brunswick Unit No. 2) as the lead plant for break spectrum and single failure analyses. We find this lead plant reference to be acceptable.

With the LPCI modification completed, the limiting break is the complete severence of the recirculation discharge line (break size of 2.366 $ft.^2$) assuming a simultaneous failure of the LPCI injection valve for the unbroken loop to open. Under such conditions, two Core Spray systems, the High Pressure Coolant Injection (HPCI) system, and the Automatic Depressurization System would be available for core cooling. This "worst-case" ECCS availability results in a faster core reflood time than that which existed for the limiting break and worst single failure for Hatch Unit No. 1 with the unmodified LPCI system. The core reflood time is improved by approximately 120 seconds for the bottom of the core and by approximately 150 seconds for the top of the core. This improved core reflood time allows an increase of approximately 10% in the maximum average planar heat generation rate limits.

For small break areas the limiting single failure continues to be failure of the HPCI system. The LPCI system modifications did not significantly affect the reflood time for these breaks.

We have reviewed the evaluation of ECCS performance with the LPCI system modifications submitted by GPC for Hatch Unit No. 1 and conclude that the evaluation is based on an acceptable evaluation model and has been performed wholly in conformance with the requirements of Section 50.46. Therefore, operation of the reactor would meet the requirements of Section 50.46 provided that operation is limited to the maximum planar linear heat generation rates (MAPLHGR) of Figure 3.11-1, sheets 1 and 2, of the GPC submittal of April 9, 1976 and to a minimum critical power ratio (MCPR) greater than 1.17. The above mentioned MAPLHGR curves appear as limiting conditions for operation in Technical Specification 3.11.A. A minimum operating MCPR limit of 1.32 is assured by Technical Specification 3.11.C.

An evaluation was not provided for ECCS performance during reactor operation with one LPCI pump or LPCI subsystem out of service. Consequently, the Technical Specifications have been modified to preclude continuous operation with a LPCI pump or LPCI subsystem inoperable, except as with other ECCS equipment one LPCI pump or LPCI subsystem may be inoperable for seven days. This change to the proposed Technical Specifications is the mutually acceptable result of discussions between the licensee and the NRC staff.

Environmental Aspects

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in authorized 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 statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change 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 change 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:

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 33 to Facility Operating License No. DPR-57 issued to Georgia Power Company and Oglethorpe Electric Membership Corporation, which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 1, located in Appling County, Georgia. The amendment is effective as of its date of issuance.

The amendment will incorporate revised operating limits into the Technical Specifications based upon an Emergency Core Cooling System (ECCS) Analysis which takes credit for the improved post-loss-of-coolant accident (LOCA) core reflood capability resulting from recently completed modifications to the Low Pressure Coolant Injection (LPCI) system.

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 §51.5(d)(4) an environmental statement, negative declaration or 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 April 9, 1976, (2) Amendment No. 33 to License No. DPR-57, 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, Parker Street, 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 Operating Reactors.

Dated at Bethesda, Maryland this day of

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors