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OCT 1 5 1976

Docket No. 50-321

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. 38 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 August 25, 1976.

The amendment revises the Hatch Unit No. 1 Technical Specifications by (1) replacing the existing Operating Limit Minimum Critical Power Ratio (MCPR) of 1.32 with exposure-dependent limits, (2) replacing the existing Safety Limit MCPR of 1.06 with a limit of 1.05, and (3) reducing the existing 100% recirculation flow Rod Block Monitor trip setting from 107% to 106% rated thermal power.

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. 38 to License No. DPR-57
2. Safety Evaluation
3. Federal Register Notice

cc:
See next page

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DATE	10/13/76	10/13/76	10/14/76	10/15/76	10/13/76

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

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. 38
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 August 25, 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;

   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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 15, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 38

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace pages 1.1-1, 1.1-10, 1.1-12 thru 1.1-14, 3.2-16, 3.2-63, 3.2-64, 3.3-5, 3.3-10, 3.11-2 and 3.11-4 thru 3.11-6 with the attached revised pages. (No change has been made on page 3.2-15).

1.1 FUEL CLADDING INTEGRITY

Applicability

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications

A. Reactor Pressure >800 psia and Core Flow > 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.05 shall constitute violation of the fuel cladding integrity safety limit.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia

When the reactor pressure is ≤ 800 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. IRM High High Flux Scram Trip Setting
The IRM flux scram trip setting shall be ≤ 120/125 of full scale.

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)
When the Mode Switch is in the REFUEL or START & HOT STANDBY position, the APRM flux scram trip setting shall be ≤ 15/125 of full scale (i.e. ≤ 15% of rated thermal power).

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the HNP-1 Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 2537 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 2436 MWt is the licensed maximum power level of HNP-1, and this represents the maximum steady-state power which shall not knowlingly be exceeded.

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

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

For analyses of the thermal consequences of the transients, MCPR is conservatively assumed to exist at the operating limit prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

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

2.1.A.1.a. IRM Flux Scram Trip Setting (Continued)

tism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is by-passed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.05. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continues withdrawal of control rods in sequence and provides backup protection for the APRM.

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, 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 to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. 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 percentage of rated power, 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 scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 880 psig.

c. APRM High High Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (2436 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

2.1,A.1.c. APRM High High Flux Scram Trip Setting (Run Mode) (Continued)

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.c, when the maximum total peaking factor is greater than 2.6.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.05 when the transient is initiated from the operating MCPR limit.

d. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.05. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during a steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds 2.6, thus preserving the APRM rod block safety margin.

2. Reactor Water Low Level Scram Trip Setting (LL1)

The trip setting for low level scram is above the bottom of the separator skirt. This level is > 14.5 feet above the top of the active fuel. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Section 14.3 show that a scram at this level adequately protects the fuel and the pressure barrier. The scram trip setting is approximately 33 inches below the normal operating range and is thus adequate to avoid spurious scrams.

2.1.A.3. Turbine Stop Valve Closure Scram Trip Settings

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.05 during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

4. Turbine Control Valve Fast Closure Scram Trip Setting

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the turbine bypass. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50% greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar and no more severe than for the stop valve. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

The main steam line isolation valve closure scram occurs within 10% of valve movement from the fully open position and thus anticipates the neutron flux and pressure scrams which remain as available backup protection. This scram function is bypassed automatically when the reactor pressure is below 1045 psig and the Mode Switch is not in the RUN position.

6. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation of the main steam lines at 880 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel, which might result from a pressure regulator failure causing inadvertent opening of the control and/or bypass valves.

Table 3.2-7

NEUTRON MONITORING INSTRUMENTATION WHICH INITIATES CONTROL ROD BLOCKS

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1	SRM	Inoperative	2(c)(d)	Not applicable	Inoperative trip produced by switch not in operate, power supply voltage low, and circuit boards not in circuit.
		Not fully inserted	2(c)(d)	Not fully inserted	This function is bypassed when the count rate is > 100 cps or the IRMs are on range 3 or above.
		Downscale	2(c)(d)	≥ 3 counts/sec	This function is bypassed when the count rate is >100 cps or the IRMs are on range 3 or above.
		Upscale	2(c)(d)	≤ 10^5 counts/sec.	
2	IRM	Inoperative	3(d)	Not applicable.	Inoperative trip produced by switch not in operate, power supply voltage low, or circuit boards not in circuit.
		Not fully inserted	3(d)	Not fully inserted	Only required in the Refuel and Start & Hot Standby Modes.
		Downscale	3(d)	≥ 5/125 of full scale	Trip bypassed when IRM on Range 1.
		High Flux	3(d)	≤108/125 of full scale	
3	APRM	Inoperative	2(e)	Not applicable	Inoperative trip produced by switch not inoperate, power supply voltage low, or circuit boards not in circuit.

Table 3.2-7 (Cont.)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
3	APRM	Downscale	2(e)	≥3/125 of full scale	Not required while performing low power physics test at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
		12% Flux	2(e)	≤12/125 of full scale	This function is bypassed when the Mode Switch is placed in the RUN position.
		High Flux	2(e)	≤0.66W + 42%	W is the loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated power. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
4	RBM	Inoperative	1(e)(f)(g)(h)	Not applicable	Inoperative trip produced by switch not in operate, circuit boards not in circuit, fails to null, less than required number of LPRM inputs for rod selected.
		Downscale	1(e)(f)(g)(h)	≥3/125 of full scale	
		High Flux	1(e)(f)(g)(h)	≤0.66W + 40%	W is the loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated thermal power.

3.2.F.5. Core Spray Pump Discharge Flow

A flow switch is provided downstream of each core spray pump to indicate the condition of each pump. To protect the pumps from overheating at low flow rates a minimum flow bypass line, which routes water from the pump discharge to the suppression chamber, is provided. A single motor-operated valve controls the condition of each bypass line. The minimum flow bypass valve automatically opens upon sensing low flow in the discharge line. The valve automatically closes whenever the flow is above the low flow setting.

6. Core Spray Pump Discharge Interlock

A pressure $\geq$ 100 psig on the core spray pump discharge indicates that the pump has started successfully. The setpoint provides a permissive signal to ADS which allows ADS initiation if other requirements are met.

7. Core Spray Logic Power Failure Monitor

The Core Spray Logic Power Failure Monitor monitors the availability of power to the logic system. In the event of loss of availability of power to the logic system, an alarm is annunciated in the control room.

G. Neutron Monitoring Instrumentation Which Initiates Control Rod Blocks (Table 3.2-7)

These control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.05. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure that the single failure criteria is met.

1. SRM

a. Inoperative

This rod block assures that no control rod is withdrawn during low neutron flux level operations unless proper neutron monitoring capability is available, in that all SRM channels are in service or properly bypassed.

b. Not Fully Inserted

Any source range monitor not fully inserted into the core when the SRM count rate level is below the retract permit level will cause a rod block. This assures that no control rod is withdrawn unless all SRM detectors are properly inserted when they must be relied upon to provide the operator with a knowledge of the neutron flux level.

c. Downscale

This rod block assures that no control rod is withdrawn unless the SRM count rate is above the minimum prescribed for low neutron flux level monitoring.

3.2.G.1.d. Upscale

This rod block assures that no control is withdrawn unless the SRM detectors are properly retracted during reactor startup. This setting is selected at the upper end of the range over which the SRM is designed to detect and measure neutron flux.

2. IRM

The trip logic for this function is 1 out of 8; any trip on one of the eight IRM's will result in a rod block. The IRM rod block function provides local as well as gross core protection.

a. Inoperative

This rod block assures that no control rod is withdrawn unless the IRM's are in service.

b. Not Fully Inserted (Refuel and Start & Hot Standby Mode)

This rod block assures that no control rod is withdrawn during low neutron flux level operations unless proper neutron monitoring capability is available in that all IRM detectors are properly located.

c. Downscale

A downscale indication of $\leq$ 5/125 full scale on an IRM is an indication that the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trip is set at $\geq$ 5/125 full scale. This rod block trip is bypassed when the IRM is on the range 1.

d. High Flux

If the IRM channels are in the worst condition of allowed bypass, the scaling arrangement is such that for unbypassed IRM channels a rod block signal is generated before the detected neutron flux has increased by more than a factor of 10.

3. APRM

The trip logic for this function is 1 out of 6; any trip on one of the six APRM's will result in a rod block. The APRM rod block function provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.05 under normal operating conditions.

a. Inoperative

This rod block assures that no control rod is withdrawn unless the APRM's are in service.

3.3.F Operation with a Limiting Control Rod Pattern

During operation with a Limiting Control Rod Pattern, either:

1. Both RBM channels shall be operable, or

2. Control rod withdrawal shall be blocked, or

3. The operating power level shall be limited so that the MCPR will remain above 1.05 assuming a single error that results in complete withdrawal of any single operable control rod.

4.3.F Operation with a Limiting Control Rod Pattern

During operation when a Limiting Control Rod Pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

G. Limiting the Worth of a Control Rod Below 20% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Whenever the reactor is in the Start & Hot Standby or Run Mode below 20% rated thermal power, the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

G. Limiting the Worth of a Control Rod Below 20% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Prior to the start of control rod withdrawal at startup, and as soon as automatic initiation of the RWM occurs during rod insertion while shutting down, the capability of the Rod Worth Minimizer to properly fulfill its function shall be verified by the following checks.

a. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.

b. The RWM computer on line diagnostic test shall be successfully performed.

c. Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

d. The rod block function of the RWM shall be verified by withdrawing or inserting an out-of-sequence control rod no more than to the block point.

3.3.C. Control Rod Drive System

1. Control Rod Drive Coupling Integrity

Limiting Conditions for Operation:

Operability of the control rod drive system requires that the drive be coupled to the control rod. In the analysis of control rod drop accidents (FSAR subsection 14.4.3), it has been assumed that one control rod drive coupling has lost its integrity. To assure that not more than one coupling could be in this condition, it is required that either a drive is coupled to the control rod or the drive is fully inserted and disarmed electrically. This requirement serves to maintain operation within the envelope of conditions considered by the plant safety analyses.

Surveillance Requirements:

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

2. Scram Insertion Times

Limiting Conditions for Operation:

The control rod drive system is designed to bring the reactor subcritical at a rate fast enough to prevent excessive fuel damage. The limiting power transient is that resulting from a loss of condenser vacuum (FSAR Appendix G, Event 12, turbine stop-valve closure with closure of the turbine bypass system). Analysis of the transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specification provide the required protection and MCPR remains greater than 1.05. The limit on the number and pattern of rods permitted to have long scram times is specified to assure that the effect of rods of long scram times are minimized in regard to reactivity insertion rate. Grouping of long scram time rods is prevented by not permitting more than one slow rod in any four rod array. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no operable control rod to have a scram insertion time for 90% insertion greater than 7 seconds.

LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.11.C. Minimum Critical Power Ratio (MCPR)

The MCPR limit is specified as a function of fuel average exposure. From 0 to 11,000 MWD/T, the MCPR limit is 1.20. From 11,000 MWD/T to end of cycle the limit is 1.24. During power operation, MCPR shall be as above at rated power and flow. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated the MCPR shall be K_f times the MCPR value applicable above, where K_f is as shown in Figure 3.11-3.

4.11.C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.F.

D. Reporting Requirements

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

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

3.11.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 integrity Safety Limit MCPR of 1.05 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 the fuel cladding integrity safety limit (MCPR of 1.05) 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.

From 0 to 11,000 MWD/T the most limiting transient was the rod withdrawal error which resulted in a Δ MCPR of 0.15. Consequently the minimum required operating limit MCPR is 1.20. The most limiting transient from 11,000 MWD/T to end of cycle was the turbine trip without bypass which resulted in a Δ MCPR of 0.19. Consequently the minimum required operating limit MCPR is 1.24.

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.

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

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.

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-21124-1, July 1976.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-321

Introduction

By letter dated August 25, 1976, Georgia Power Company (GPC) requested an amendment to Facility Operating License No. DPR-57 for Edwin I. Hatch Nuclear Plant Unit No. 1 (HNP-1). The proposed amendment would modify the HNP-1 Technical Specifications by:

a. Replacing the existing Operating Limit Minimum Critical Power Ratio (MCPR) of 1.32 with exposure-dependent Operating Limit MCPRs of 1.20 for fuel exposures up to 11,000 Megawatt-days/Ton (MWD/T) and 1.24 for fuel exposures from 11,000 MDW/T to the end of fuel cycle 1 (EOC-1);

b. Replacing the existing Safety Limit MCPR of 1.06 with a limit of 1.05; and

c. Replacing the existing Rod Block Monitor trip setting of "$\leq$0.66W + 41%" with a setting of "$\leq$0.66W + 40%" (i.e., changing the full recirculation flow setting from 107% to 106% rated thermal power.

Discussion

On December 17, 1975 the NRC issued License Amendment No. 27 for HNP-1 which authorized operation with a Safety Limit MCPR of 1.06 and with an Operating Limit MCPR of 1.32. These limits were based on a thermal-hydraulic analysis[1] of the HNP-1 core using the General Electric BWR Thermal Analysis Basis (GETAB). The analysis was performed using (1) the EOC-1 scram reactivity insertion rate which conservatively bounds the actual exposure-dependent scram reactivity insertion rates for the entire fuel cycle, and (2) an analytically-determined value for fuel bundle bypass flow for the plugged lower core support plate configuration.

In order to provide greater operating flexibility throughout fuel cycle 1, GPC has proposed to incorporate revised operating limits based upon a thermal-hydraulic performance reanalysis which (1) includes the effects of exposure-dependent scram reactivity curves, and (2) utilizes new bypass flow data which has been obtained from a full scale test facility.

Evaluation

Revised Bypass Flow Information

During the conduct of the full scale testing program[2] associated with the development of permanent plant modifications to eliminate the recurrence of significant channel box wear in the GE BWR-4 product line (which includes HNP-1), it was experimentally demonstrated that the total fuel bundle bypass flow for a plugged core configuration is greater than that which has been predicted previously using analytical models. For HNP-1 in the plugged configuration, the experimentally derived bypass flow is approximately 6.4% of total core flow, whereas the analytical calculation had previously predicted a bypass flow of approximately 4.8% of total core flow. Application of this new bypass flow information to HNP-1 reveals that there is actually less void formation in the bypass region of the core during plant operation than that which had been assumed in the previous thermal-hydraulic analysis. This reduction in the amount of bypass-region void formation is reflected in the HNP-1 thermal-hydraulic reanalysis by (1) utilization of a less negative value for the void coefficient of reactivity in the transient analysis, (2) a revision to the EOC-1 scram reactivity curve, and (3) a reduction in the Traversing In-Core Probe (TIP) reading uncertainty value used in the determination of the Safety Limit MCPR.

We have reviewed the new bypass flow information derived from the experimental testing program and have concluded that its application to the HNP-1 thermal-hydraulic reanalysis is acceptable.

Safety Limit MCPR

The Safety Limit MCPR is the value of the critical power ratio (CPR) at which 99.9% of the fuel rods in the core would be expected to avoid boiling transition. Boiling transition represents an unstable heat transfer condition on the fuel clad surface and could result in fuel clad temperatures exceeding allowable limits. The Safety Limit MCPR provides an upper bound or limit which will assure fuel cladding integrity during any anticipated abnormal operational transient.

The proposed change in the fuel cladding integrity Safety Limit MCPR from 1.06 to 1.05 reflects the decrease in the uncertainties associated with the TIP system readings resulting from application of the new bypass flow data to HNP-1. The Safety Limit MCPR value is determined by means of a GETAB statistical analysis which considers the relative power distribution in the core, the uncertainties associated with the GEXL correlation, and the uncertainties associated with reactor steady-state operating parameters. We have previously reviewed and approved GETAB and GEXL.

We have reviewed the information related to the decrease in the uncertainties associated with the TIP system readings and have concluded that the associated change to the Safety Limit MCPR is acceptable.

Operating Limit MCPR

Various transient events will reduce the CPR below its steady-state operating value. To assure that the Safety Limit MCPR of 1.05 is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which transient results in the largest reduction in critical power ratio (ΔCPR). Subsequent addition of the largest value of CPR to the Safety Limit MCPR then establishes the minimum value of the operating limit CPR which assures that the Safety Limit MCPR and thus the fuel cladding integrity will not be violated should the most limiting abnormal operational transient occur.

GPC has submitted the results of an analysis of those abnormal operational transients which could cause a significant decrease in CPR. The types of transients evaluated were loss of flow, pressure and power increase, coolant temperature decrease and rod withdrawal error. The most limiting transients in the stated categories are as follows:

Event	ΔCPR 105% Rated Power 0 to 11,000 MWD/T	ΔCPR 105% Rated Power 11,000 MWD/T to EOC-1
Turbine Trip w/o bypass	0.06	0.19
Loss of Feedwater Heater	0.14	0.14
Rod Withdrawal Error	0.15	0.15

As shown above, the Rod Withdrawl Error transient is the most severe abnormal operational transient from the beginning of the fuel cycle 1 through a fuel exposure of 11,000 MWD/T, at which time the Turbine Trip without bypass transient becomes most limiting. The resulting required Operating Limit MCPR values which assure that the Safety Limit MCPR will not be violated are 1.20 for 0 to 11,000 MWD/T exposure and 1.24 for 11,000 MWD/T exposure to EOC-1.

The transient analyses for the two analyzed portions of the fuel cycle 1 were evaluated with scram reactivity insertion rate curves for 11,000 MWD/T exposure and EOC-1 exposure respectively. In both cases the scram reactivity insertion rate curve used in the analyses included a design conservation factor and conservatively bounded the exposure-dependent scram reactivity insertion rates for the applicable portion of the fuel cycle. The initial parameters used for the transient analyses were acceptable and included initial CPR values which were equal to or greater than the established Operating Limit MCPR values for each portion of the fuel cycle. The previously approved ECCS analysis[3] assumed that reactor operation will be limited to an operating CPR no lower than 1.17. The proposed Operating Limit MCPR values are higher than 1.17 and are acceptable in terms of ECCS performance.

Based on the discussion above, we conclude that the proposed changes to the Operating Limit MCPR values are acceptable.

Rod Withdrawal Error Transient

GPC discussed the rod withdrawal error transient in terms of worst case conditions. The analysis shows that the local power range monitor subsystem (LPRM's) will detect high local power in the core and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop rod withdrawal before either the Safety Limit MCPR of 1.05 or the fuel cladding plastic strain limit of 1% are exceeded. This analysis assumed that the RBM trip setting for full recirculation flow is 106% rated thermal power; consequently, the HNP-1 Technical Specifications have been revised to assure that facility operation will be in conformance with this assumption. We conclude that the consequences of this localized transient are acceptable.

Overpressurization Protection

GPC's previous submittal[1] on overpressure protection was based upon EOC-1 conditions. It showed that there was adequate margin to meet the ASME code allowable pressure in the event of MSIV closure with flux scram, no relief function of the safety relief valves and one failed safety valve. That analysis may be conservatively applied throughout fuel cycle 1 since it was performed for the most severe conditions (EOC-1)

using a larger negative void coefficient and a lesser scram worth effectiveness factor than that used in the current analysis. Hence, as in Reference 4, we find the overpressure protection to be acceptable.

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 changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do 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: October 15, 1976

REFERENCES

1. "Edwin I. Hatch Nuclear Plant Unit No. 1 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged", NEDO-21124, November 1975.

2. "Supplemental Information For Plant Modification to Eliminate Significant In-Core Vibration", NEDE-21156, January 1976.

3. "Edwin I. Hatch Nuclear Plant Unit No. 1 Emergency Core Cooling System Analysis - Appendix K Requirements With Modified Low Pressure Coolant Injection System", NEDO-21187, February 1976; and Supplement No. 1, April 1976.

4. Safety Evaluation by the Office of Nuclear Reactor Regulation of Amendment No. 27 to Facility Operating License No. DPR-57, December 17, 1975.

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

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 38 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 revises the Hatch Unit No. 1 Technical Specifications by (1) replacing the existing Operating Limit Minimum Critical Power Ratio (MCPR) of 1.32 with exposure-dependent limits, (2) replacing the existing Safety Limit MCPR of 1.06 with a limit of 1.05, and (3) reducing the existing 100% recirculation flow Rod Block Monitor trip setting from 107% to 106% rated thermal 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 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 August 25, 1976, (2) Amendment No. 38 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 15 day of October 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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