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Docket No. 50-321

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Mr. Charles F. Whitmer
Vice President - Engineering
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Whitmer:

The Commission has issued the enclosed Amendment No. 69 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 to: (1) permit operation of the facility during Cycle 4 with 164 reload fuel assemblies of the GE 8x8 retrofit design, and (2) permit modification of the APRM trip system by incorporating a Thermal Power Monitor. The amendment is in partial response to your application dated March 22, 1979 as supplemented by letters dated May 11 and 16, 1979. We have not yet completed our review of the hardware implementation of the End-of-Cycle (EOC) Recirculation Pump Trip features. Therefore, in order not to delay the restart of the facility upon completion of the current refueling outage, we recommended to your staff that the amendment contain a restriction that the Operating Limit Minimum Critical Power Ratios (MCPR) be effective only for Cycle 4 operation up to 2000 MWd/t before EOC. This is a conservative limit as discussed in our evaluation. Your staff agreed.

Copies of our Safety Evaluation and a related Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

CP

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 69 to DPR-57
2. Safety Evaluation
3. Notice

*SEE PREVIOUS YELLOW FOR CONCURRENCES

ORB #3	ORB #3
PKreutzer	*DVerrelli:mjf
8/6/79	7/26/79

GD

OFFICE	cc w/enclosures:	AD:ORP	ORB #3	RSB	PSB	OELD
SURNAME	See page 2	*WGammill	*TIppolito	*PCheck	*GLainas	*
DATE		7/31/79	7/31/79	7/26/79	7/31/79	8/3/79

Docket No. 50-321

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See page 2

OFFICE ➤		AD:ORR	ORB #3	ORB #3	ORB #3	RSB	PSB	OELD
SURNAME ➤		WGammill	TIppolito	PKreutzer	DVerrelli:mjf	PCheck	GLainas	
DATE ➤		7/3/79	7/31/79	7/ /79	7/26/79	7/26/79	7/3 /79	8/3/79

NRC FORM 318 (9-76) NRCM 0240

cc:

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

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

Mr. Harry Majors
Southern Services, Inc.
300 Office Park
Birmingham, Alabama 35202

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

Mr. H. B. Lee, Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

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

Mr. William Widner
Georgia Power Company
Power Generation Department
P. O. Box 4545
Atlanta, Georgia 30302

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

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

Appling County Public Library
Parker Street
Baxley, Georgia 31513

Mr. R. F. Rodgers
U. S. Nuclear Regulatory Commission
P. O. Box 710
Baxley, Georgia 31513

Director, Technical Assessment
Division
Office of Radiation Programs (AW 459)
US EPA
Crystal Mall #2
Arlington, Virginia 20460

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

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION
MUNICIPAL ELECTRIC ASSOCIATION OF GEORGIA
CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company et al., (the licensee) dated March 22, 1979 as amended May 11 and 16, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C. (2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 69, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 6, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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

REMOVE	INSERT
ix	ix
1.1-1	1.1-1
1.1-2	1.1-2
1.1-12	1.1-12
1.1-13	1.1-13
Fig. 1.1-1	Fig. 1.1-1
1.2-3	1.2-3
1.2-5	1.2-5
3.1-4	3.1-4
3.1-7	3.1-7
3.1-11 *	3.1-11
3.1-12	3.1-12
3.2-27	3.2-27
3.2-28 *	3.2-28
3.2-29 *	3.2-29
3.2-30	3.2-30
3.6-20	3.6-20
3.11-1	3.11-1
3.11-2	3.11-2
3.11-4	3.11-4
Fig. 3.11-2 (Sheet 1)	(deleted)
Fig. 3.11-2 (Sheet 2)	(deleted)
5.0-1	5.0-1

* Overleaf

LIST OF FIGURES

Figure	Title
1.1-1	Core Thermal Power Safety Limit Versus Core Flow Rate
2.1-1	Reactor Vessel Water Levels
4.1-1	Graphical Aid for the Selection of an Adequate Interval Between Tests
4.2-1	System Unavailability
3.4-1	Sodium Pentaborate Solution Volume Versus Concentration Requirements
3.4-2	Sodium Pentaborate Solution Temperature Versus Concentration Requirements
3.6-1	Change in Charpy V Transition Temperature Versus Neutron Exposure
3.6-2	Minimum Temperature for Inservice Hydrostatic and Leak Tests
3.6-3	Minimum Temperature for Mechanical Heatup or Cooldown Following Nuclear Shutdown
3.6-4	Minimum Temperature for Core Operation (Criticality)
3.11-1	(Sheet 1) Limiting Value for APLHGR (Fuel Type 3)
3.11-1	(Sheet 2) Limiting Value for APLHGR (Fuel Types 1 and 2)
3.11-2	deleted
3.11-3	K_f Factor
6.2.1-1	Offsite Organization
6.2.2-1	Unit Organization

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 1.07 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 psia 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).

c. APRM Flux Scram Trip Setting (Run Mode)

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position the APRM flow referenced flux scram trip setting shall be:

Amendment No. 27, 38, 42, 52, 69

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

Whenever the reactor is in the Hot or Cold Shutdown Condition with irradiated fuel in the reactor vessel, the water level shall be > 378 inches above vessel invert when fuel is seated in the core.

2.1.A.1.C APRM Flux Scram Trip Settings (Run Mode) (Continued)

$$S \leq 0.66\ W + 54\%$$

where:

S = Setting in percent of rated thermal power (2436 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value, the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%)\ \frac{A}{MTPF}$$

where:

MTPF = The value of the existing maximum total peaking factor

A = 2.60 for 7x7 fuel
2.42 for 8x8 fuel
2.48 for 8x8R fuel

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

Surveillance requirements for MTPF are given in Specification 4.1.B.

(2) Fixed High Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

$$S \leq 120\%\ \text{Power}$$

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 bypassed. 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 1.07. 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 accomodate 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 825 psig.

c. APRM Flux Scram Trip Settings (Run Mode)

The APRM Flux scram trip in the run mode consists of a flow referenced scram setpoint and a fixed high 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. This prevents spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Examples of events which can result in momentary neutron flux spikes are momentary flow changes in the recirculation system flow, and small pressure disturbances during turbine stop valve and turbine control valve testing. These flux spikes represent no hazard to the fuel since they are only of a few seconds duration and less than 120% of rated thermal power.

2.1.A.1:c. APRM Flux Scram Trip Settings (Run Mode) (Continued)

The APRM flow referenced scram trip setting at full recirculation flow is adjustable up to 117% of rated power. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 80^{o}F feedwater heating event, than would result with the 120% fixed high 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 cycle.

The APRM fixed high 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 scram.

The flow referenced 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 adusted in accordance with the formula in Specification 2.1.A.1.c., when the maximum total peaking factor is greater than 2.60 for 7x7 fuel, 2.42 for 8x8 fuel and 2.48 for 8x8R fuel.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 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 1.07. 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 would 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 2.60 for 7x7 fuel, 2.42 for 8x8 fuel and 2.48 for 8x8R fuel, 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 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.

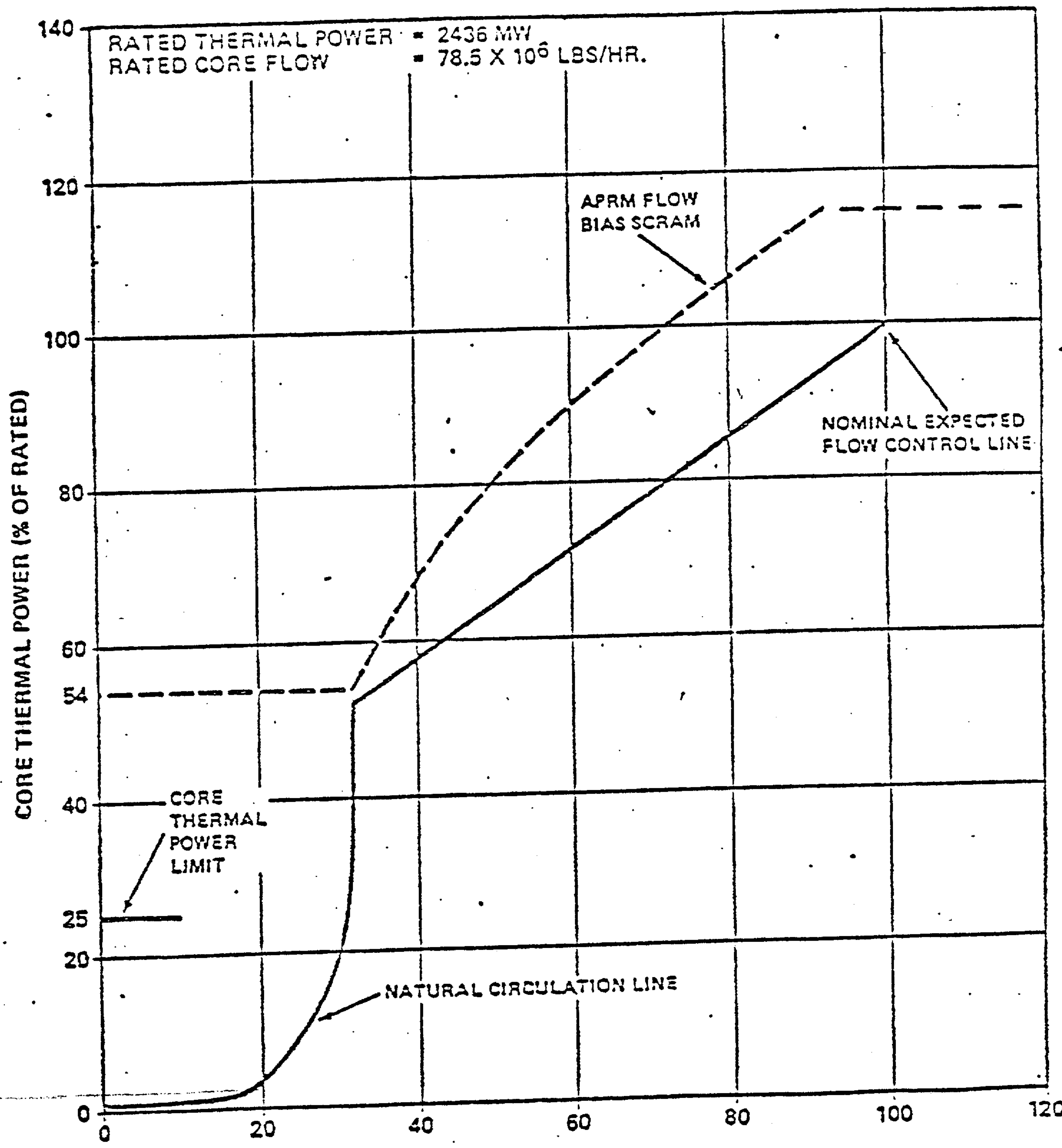

FIGURE 1.1-1
CORE THERMAL POWER SAFETY LIMIT
VERSUS CORE FLOW RATE

1.2 REACTOR COOLANT SYSTEM INTEGRITY

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

A. Reactor Vessel Steam Dome Pressure

1. When Irradiated Fuel is in the Reactor

The pressure Safety Limit of 1325 psig as measured by the reactor vessel steam dome pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressure of the reactor pressure vessel (1250 psig) and coolant system piping (suction piping: 1150 psig; discharge piping: 1350 psig). The pressure Safety Limit was chosen as the lower pressure resulting from the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1150 = 1380 psig; 120% x 1350 = 1602 psig).

The pressure relief system (relief/safety valves) has been sized to meet the overpressure protection criteria of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

The details of the overpressure protection analysis showing compliance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels is provided in the FSAR, Appendix M, Summary Technical Report of Reactor Vessel Overpressure Protection. To determine the required steamflow capacity, a parametric study was performed assuming the plant was operating at the turbine generator design condition of 105 percent rated steam flow (10.6 x 10^6 pounds per hour) with a vessel dome pressure of 1020 psig, at a reactor thermal power of 2537 Mw, and the reactor experiences the worst pressurization transient. The analysis of the worst overpressure transient, a 3 second closure of all main steam line isolation valves neglecting the direct scram (valve position scram) results in a maximum vessel pressure (bottom) of less than 1375 psig if a neutron flux scram is assumed. In addition, the same event was analyzed to determine the number of installed valves which would limit pressure to below the code limit. The results of this analysis show that the eleven installed relief/safety valves were adequate even if assuming the backup neutron flux scram.

Turbine trip from high power without bypass is the most severe transient resulting directly in a nuclear system pressure increase, assuming direct scram. This event is presented in Reference 5. The analysis shows that the peak pressure in the bottom of the vessel is limited to 1180 psig. Peak steam line pressure is 1149 psig, showing adequate protection for this abnormal operational transient.

1.2.B. References

1. ASME Boiler and Pressure Vessel Code Section III.

2. USASI Piping Code, Section B31.1.

3. FSAR Section 4.2, Reactor Vessel and Appurtenances Mechanical Design.

4. FSAR Section 14.3, Analysis of Abnormal Operation Transients.

5. General Electric Boiling Water Reactor Supplemental Reload Licensing Submittal for the Edwin I. Hatch Nuclear Plant Unit 1 Reload 3, NEDO-24175, January, 1979.

Table 3.1-1 (Cont'd)

Scram Number (a)	Source of Scram Trip Signal	Operable Channels Required Per Trip System (b)	Scram Trip Setting	Source of Scram Signal is Required to be Operable Except as Indicated Below
5	High Drywell Pressure	2	$\leq$ 2 psig	Not required to be operable when primary containment integrity is not required. May be bypassed when necessary during purging for containment inerting or deinerting.
6	Reactor Water Low Level (LL1) (Narrow Range)	2	$\geq$ 12.5 inches	
7	Scram Discharge Volume High High Level	2	$\leq$ 71 gallons	Permissible to bypass (initiates control rod block) in order to reset RPS when the Mode Switch is in the REFUEL or SHUTDOWN position.
8	APRM Flow Referenced Neutron Flux	2	S $\leq$ 0.66W+54% (Not to exceed 117%) Tech Spec 2.1.A.1.c	
	Fixed High Neutron Flux	2	S $\leq$ 120% Power Tech Spec 2.1.A.1.c	
	Inoperative	2	Not Applicable	An APRM is inoperable if there are less than two LPRM inputs per level or there are less than 11 LPRM inputs to the APRM channel

Amendment No. ~~20~~, 69

Table 4.1-1

Reactor Protection System (RPS) Instrumentation Functional Test, Functional Test Minimum Frequency, and Calibration Minimum Frequency

Scram Number (a)	Source of Scram Trip Signal	Group (b)	Instrument Functional Test Minimum Frequency (c)	Instrument Calibration Minimum Frequency
1	Mode Switch in SHUTDOWN	A	Once/Operating Cycle	Not Applicable
2	Manual Scram	A	Every 3 months	Not Applicable
3	IRM High High Flux	C	Once/Week during refueling and within 24 hours of Startup (e)	Once/Week
	Inoperative	C	Once/week during refueling and within 24 hours of Startup (e)	Once/Week
4	High Reactor Pressure	A	Once/Month (f)	Every 3 months
5	High Drywell Pressure	A	Once/Month (f)	Every 3 months
6	Reactor Water Low Level (LL1)	A	Once/Month (f) (g)	Every 3 months
7	Scram Discharge Volume High High Level	A	Every 3 months	(h)
8	APRM Fixed High Flux	B	Once/Week (e)	Twice/Week
	Inoperable	B	Once/Week (e)	Twice/Week
	Downscale	B	Once/Week (e)	Twice/Week
	Flow Reference	B	Once/Week (f)	Once/Operating Cycle
	15% Flux	C	Within 24 Hours of Startup (e)	Once/Week

3.1.A.2. Manual Scram

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

3. IRM

The bases for the IRM High High Flux Scram Trip Setting are discussed in the bases for Specification 2.1.A.1.a. Each protection trip system has one more IRM channel than is necessary to meet the minimum number required. This allows the bypassing of one IRM channel per protection trip system for maintenance, testing or calibration.

a. High High Flux

The IRM system provides proteetion against excessive power levels and short reactor periods in the source and intermediate power ranges. The requirement that the IRM's be inserted in the core until the APRM's read 3/125 of full scale or greater assures that there is proper overlap in the neutron monitoring systems and thus, that adequate coverage is provided for all ranges of reactor operation.

A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram function (Section 7.5.4 FSAR). Thus, the IRM and APRM systems are required in the Refuel and Start & Hot Standby modes. In the power range, the APRM system provides the required protection (Section 7.5.7 FSAR). Thus, the IRM System is not required when the APRM's are on scale and the Mode Switch is in the RUN position.

b. Inoperative

When an IRM channel becomes unable to perform its normal monitoring function, the condition is recognized and an inoperative trip results. This trip is given the same logic significance as the upscale trip; thus the faulty channel immediately fails safe by contributing to a potential scram condition.

4. High Reactor Pressure

High pressure within the nuclear system poses a direct threat of rupture to the nuclear system process barrier. A nuclear system pressure increase while the reactor is operating compresses the steam voids and results in a positive reactivity insertion causing increased core heat generation that could lead to fuel failure and system over-pressurization. A scram counteracts a pressure increase by quickly reducing the core fission heat generation.

The nuclear system high pressure scram setting is chosen slightly above the reactor vessel maximum normal operation pressure to permit normal operation without spurious scrams yet provide a wide margin to the maximum allowable nuclear system pressure. The location of the pressure measurement, as compared to the location of highest nuclear system pressure during transients, was also considered in the selection of the high pressure scram setting. The nuclear system high pressure scram works in conjunction with the pressure relief system in preventing nuclear system pressure from exceeding the maximum allowable pressure. This same nuclear system high pressure scram setting also protects the core from exceeding thermal hydraulic limits as a result of pressure increases for some events that occur when the reactor is operating at less than rated power and flow.

5. High Drywell Pressure

Pressure switch instrumentation for the drywell is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting ($\leq$2 psig) as the core standby cooling systems initiation to minimize the energy which must be accommodated during a loss of coolant accident. The instrumentation is a backup to the reactor vessel water level instrumentation.

6. Reactor Water Low Level (LL1)

The bases for the Reactor Water Low Level Scram Trip Setting (LL1) are discussed in the bases for Specification 2.1.A.2.

7. Scram Discharge Volume High High Level

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this piping is an instrument volume which is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should the discharge volume fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in a slow scram time or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which scram the reactor when the volume of water reaches 71 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not able to perform its function adequately.

8. APRM

Three APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one trip logic and APRM's C and E operate contacts in the other trip logic. APRM's B, D and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration.

a. Flow Referenced and Fixed High Neutron Flux

The bases for the APRM Flow Referenced and Fixed High Neutron Flux Scram Trip Settings are discussed in the bases for Specification 2.1.A.1.c.

Table 4.2-2

Check, Functional Test, and Calibration Minimum Frequency for Instrumentation Which Initiates or Controls HPCI

Ref. No. (a)	Instrument	Instrument Check Minimum Frequency	Instrument Functional Test Minimum Frequency (b)	Instrument Calibration Minimum Frequency (c)
1	Reactor Water Level (Yarway)	Once/day	(d)	Every 3 months
2	Drywell Pressure	None	(d)	Every 3 months
3	HPCI Turbine Overspeed	None	N/A	Once/operating cycle
4	HPCI Turbine Exhaust Pressure	None	(d)	Every 3 months
5	HPCI Pump Suction Pressure	None	(d)	Every 3 months
6	Reactor Water Level (Narrow Range)	Once/day	(d)	Every 3 months
7	HPCI System Flow (Flow Switch)	None	(d)	Every 3 months
8	HPCI Equipment Room Temperature	None	(d)	Every 3 months
9	deleted			
10	HPCI Steam Line Pressure	None	(d)	Every 3 months

Table 4.2-2 (Cont'd)

Ref. No. (a)	Instrument	Instrument Check Minimum Frequency	Instrument Functional Test Minimum Frequency (b)	Instrument Calibration Minimum Frequency (c)
11	HPCI Steam Line ΔP (Flow)	None	(d)	Every 3 months
12	HPCI Turbine Exhaust Diaphragm Pressure	None	(d)	Every 3 months
13	Suppression Chamber Area Air Temperature	None	(d)	Every 3 months
14	Suppression Chamber Area Differential Air Temperature	None	(d)	Every 3 months
15	Condensate Storage Tank Level	None	(d)	Every 3 months
16	Suppression Chamber Water Level	None	(d)	Every 3 months
17	HPCI Logic Power Failure Motor	None	Once/operating cycle	None

Notes for Table 4.2-2

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 4.2-2 and items in Table 3.2-2.

Notes for Table 4.2-2 (Cont'd)

b. Instrument functional tests are not required when the instruments are not required to be operable or are tripped. However, if functional tests are missed, they shall be performed prior to returning the instrument to an operable status.

c. Calibrations are not required when the instruments are not required to be operable. However, if calibrations are missed, they shall be performed prior to returning the instrument to an operable status.

d. Initially once per month or according to Figure 4.1-1 with an interval of not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other BWR's for which the same design instrument operates in an environment similar to that of HNP-1. The failure rate data must be reviewed and approved by the AEC prior to any change in the once-a-month frequency.

Logic system functional tests and simulated automatic actuation shall be performed once each operating cycle for the following:

1. HPCI Subsystem
2. HPCI Subsystem Auto Isolation
3. Diesel Generator Initiation
4. Area Cooling for Engineered Safeguard Systems

The logic system functional tests shall include a calibration of time relays and timers necessary for proper functioning of the trip systems.

Table 4.2-3

Check, Functional Test, and Calibration Minimum Frequency for Instrumentation Which Initiates or Controls RCIC

Ref. No. (a)	Instrument	Instrument Check Minimum Frequency	Instrument Functional Test Minimum Frequency (b)	Instrument Calibration Minimum Frequency (c)
1	Reactor Water Level (Yarway)	Once/day	(d)	Every 3 months
2	RCIC Turbine Overspeed			
	Electrical/	None	N/A	Once/operating cycle
	Mechanical	None	N/A	Once/operating cycle
3	RCIC Turbine Exhaust Pressure	None	(d)	Every 3 months
4	RCIC Pump Suction Pressure	None	(d)	Every 3 months
5	Reactor Water Level (Narrow Range)	Once/day	(d)	Every 3 months
6	RCIC System Flow (Flow Switch)	None	(d)	Every 3 months
7	RCIC Equipment Room Temperature	None	(d)	Every 3 months
8	deleted			
9	RCIC Steam Line Pressure	None	(d)	Every 3 months

3.6.G. Reactor Coolant Leakage (Continued)

would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in Specification 3.6.G on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in Specification 3.6.G, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation (Reference FSAR, Question 10.4.2). Leakage less than the magnitude specified can be detected reasonably in a manner of a few hours utilizing the available leakage detection scheme, and if the origin cannot be determined in a reasonably short time the plant shall be shutdown to allow further investigation and corrective action. The total leakage rate consists of all leakage, identified and unidentified which flows to the drywell floor drain and equipment drain sump. The capacity of the drywell floor sump pumps is 100 gpm and the capacity of the drywell equipment sump pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

H. Relief/Safety Valves

The pressure relief system (relief/safety valves) has been sized to meet the overpressure protection criteria of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

The details of the overpressure protection analysis showing compliance with ASME, Section XII is provided in the FSAR, Appendix M, Summary Technical Report of Reactor Vessel Overpressure Protection. To determine the required steamflow capacity, a parametric study was performed assuming the plant was operating at the turbine-generator design condition of 105 percent rated steam flow (10.6 x 10^6 pounds per hour) with a vessel dome pressure of 1020 psig, at a reactor thermal power of 2537 Mw, and the reactor experiences the worst pressurization transient. The reanalysis for Reload-3 (NEDO-24175) of the worst overpressure transient, a 3 second closure of all main steam line isolation valves neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1232 psig if a reutron flux scram is assumed.

Turbine trip from high power without bypass is the most severe transient resulting directly in a nuclear system pressure increase, assuming direct scram. This event is presented in NEDO-24175. The analysis shows that the peak pressure in the bottom of the vessel is limited to 1180 psig. Peak steam line pressure is 1149 psig, showing adequate protection for this worst abnormal operational transient.

3.11 FUEL RODS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.11-1, sheets 1 and 2. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

B. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR as a function of core height shall not exceed the limiting value shown in Figure 3.11-2 for 7 x 7 fuel or the limiting value of 13.4 kw/ft for 8 x 8/ 8 x 8R fuel. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the

4.11 FUEL RODS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥ 25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

The LHGR as function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.

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

LHGR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

C. Minimum Critical Power Ratio (MCPR)

The MCPR limit is specified throughout the cycle. From BOC4 to EOC4-2000 MWD/t the MCPR limit is 1.26 for 7 x 7, 1.24 for 8 x 8, and 1.21 for 8 x 8R fuels.* 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, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the Limiting Condition for Operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required. 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.

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.

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.

*MCPR values for EOC4-2000 MWD/t to EOC4 will be determined after completion of the review of the hardware implementation of EOC Recirculation Pump Trip feature.

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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 for 7 x 7 fuel 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 1.07 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 1.07) 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 6-2 of Reference 9 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 BOC4 to EOC4, the most limiting transient for the 8 x 8R fuel is the loss of 100°F feedwater heating with a ΔCPR of 0.14. The most limiting event throughout cycle 4 for 8 x 8 and 7 x 7 fuel is the Rod Withdrawal Error (RWE) with a ΔCPR of 0.17 for 8 x 8 and 0.19 for 7 x 7. Therefore, the MCPR's specified in 3.11.C are based on loss of 100°F feedwater heating and the Rod Withdrawal Error.

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A. Site

Edwin I. Hatch Nuclear Plant Unit No. 1 is located on a site of about 2244 acres, which is owned by Georgia Power Company, on the south side of the Altamaha River in Appling County near Baxley, Georgia. The Universal Transverse Mercator Coordinates of the center of the reactor building are: Zone 17R LF 372,935.2m E and 3,533,765.2m N.

B. Reactor Core

1. Fuel Assemblies

The core shall consist of not more than 560 fuel assemblies of the licensed combination of 7 x 7 bundles which contain 49 fuel rods and 8 x 8 and 8 x 8R fuel bundles which contain 62 or 63 fuel rods each.

2. Control Rods

The reactor shall contain 137 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of its theoretical density.

C. Reactor Vessel

The reactor vessel is described in Table 4.2-2 of the FSAR. The applicable design specifications shall be as listed in Table 4.2-1 of the FSAR.

D. Containment

1. Primary Containment

The principal design parameters and characteristics of the primary containment shall be as given in Table 5.2-1 of the FSAR.

2. Secondary Containment

The secondary containment shall be as described in Section 5.3.3.1 of the FSAR and the applicable codes shall be as given in Section 12.4.4 of the FSAR.

3. Primary Containment Penetrations

Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.3.4 of the FSAR.

E. Fuel Storage

1. Spent Fuel

All arrangements of fuel in the spent fuel storage racks shall be maintained in a subcritical configuration having a k_{eff} not greater than 0.90 for normal conditions and a k_{eff} not greater than 0.95 for abnormal conditions.

2. New Fuel

The new fuel storage vault shall be such that the k_{eff} dry shall not be greater than 0.90 and the k_{eff} flooded shall not be greater than 0.95.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION
MUNICIPAL ELECTRIC ASSOCIATION OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-321

1.0 Introduction

Georgia Power Company (GPC) has proposed changes to the Technical Specifications of the Edwin I. Hatch Nuclear Plant, Unit 1.(1) The proposed changes relate to the replacement of 164 fuel assemblies constituting refueling of the core for Cycle 4 operation at power levels up to 2436 Mwt (100% power).

In support of the reload application, the licensee has provided proposed Technical Specification changes(2) and the GE BWR supplemental licensing submittal for Hatch-1.(3)

This reload involves loading of GE 8x8 retrofit (8x8R) fuel. The description of the nuclear and mechanical designs is contained in References 4 and 5. Reference 4 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing a mixture of 8x8 and 8x8R fuel.

Values for plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other design parameters are provided in Reference 4. Additional plant and cycle dependent information is provided in the reload application (Reference 3) which closely follows the outline of Appendix A of Reference 4.

Appendix C of Reference 4 includes a description of the staff's review, approval, and conditions of approval for the plant-specific data addressed in Reference 4. The above-mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

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Our safety evaluation (Reference 4) of the GE generic reload licensing topical report has also concluded that the nuclear and mechanical design of the 8x8R fuel, and GE's analytical methods for nuclear and thermal-hydraulic calculations as applied to mixed cores containing 8x8 and 8x8R fuel, are acceptable. Approval of the application of the analytical methods did not include plants incorporating a prompt recirculation pump trip (RPT) or Thermal Power Monitor (TPM).

Because of our review of a large number of generic considerations related to use of 8x8R fuel in mixed loadings, and on the basis of the evaluations which have been presented in Reference 4, only a limited number of additional areas of review have been included in this safety evaluation report. For evaluations of areas not specifically addressed in this safety evaluation report, the reader is referred to Reference 4.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 4 operation of Hatch-1, 164 fresh 8x8R fuel bundles of type 8DRB265H will be loaded into the core.[3] The remainder of the 560 fuel bundles in the core will be 168 once-burned type 8DRB265H bundles, 92 twice-burned type 8DB250 bundles, and 136 bundles from the initial core.

The fresh fuel will be loaded and the previously peripheral fuel will be shuffled inward to constitute an octant-symmetric core pattern.

Based on the data provided in Reference 3, both the control rod system and the standby liquid control system will have acceptable shutdown capability during Cycle 4.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 4, for BWR cores which reload with GE's retrofit 8x8R fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 SLMCPR to be used for Cycle 4 is unchanged from the SLMCPR previously approved for Cycle 3. The basis for this safety limit is addressed in Reference 4, while our generic approval of the limit is given in the staff evaluation included in Reference 4.

2.2.2 Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for the exposed 7x7 and 8x8 fuel and the exposed and fresh 8x8R fuel. Addition of the largest reductions in critical power ratio to the SLMCPR establishes the operating limits for each fuel type.

2.2.2.1 Transient Analysis Methods

The generic methods used for these calculations, including cycle-independent initial conditions and transient input parameters, are described in Reference 4. The staff evaluation, included as Appendix C of Reference 4, contains our acceptance of the cycle-independent values. Additionally, Appendix C contains our evaluation of the transient analysis methods, together with a description and summary of the outstanding issues associated with these methods. Supplementary cycle-independent initial conditions and transient input parameters used in the transient analyses appear in the tables in Sections 6 and 7 of Reference 3. Our evaluation of the methods used to develop these supplementary input values is also included in Appendix C of Reference 4.

At the time we completed our evaluation of the generic methods, the acceptability of the GEXL critical power correlation for use in connection with the retrofit fuel design had not been adequately documented by GE. The staff found, however, that the then available 8x8R critical power test data was sufficient to support the acceptability of GE's retrofit 8x8 fuel design for BWR core reloads for one operating cycle. Accordingly, we stated [4] that future BWR core reload applications involving retrofit 8x8 fuel for a second operating cycle would have to include additional information which adequately justified the correlation for application to 8x8R fuel operating beyond one cycle.

GE has prepared a report on this subject [7] that provides the results of full scale critical power tests performed on 8x8R fuel bundles. The tests included both transient and steady-state simulations and followed the same approved procedures [6] used for the standard 8x8 and 7x7 fuel designs. The analysis of a total of 577 steady-state data points was performed using methods also previously approved by the staff. The data spanned a range of local power peaking and flow conditions. GE stated that the GEXL correlation was applicable to the retrofit fuel after adjustments were made to the additive constants used in the formulation of the rod-by-rod R-factors (Figure 3-1 of Reference 7).

Using the new additive constants, GE assessed the accuracy and precision of the GEXL correlation. The results showed that the correlation provides for a mean predicted-to-measured critical power ratio of 0.9879 with a standard deviation of 0.0234.

The 8x8R GEXL correlation has a conservative bias when viewed over the entire range of its applicability (which is the same as the 8x8 correlation). Thus, the 8x8R GEXL correlation has better precision than the 7x7 and 8x8 GEXL correlations for predicting critical bundle powers when viewed over the entire range of applicability. Furthermore, the 3.6% standard deviation and zero bias of the GEXL correlation bound the statistical characteristics of the 8x8R GEXL correlation used in the GETAB statistical analysis to derive the 1.07 safety limit MCPR.

The information furnished by GE [7] was intended to apply to all BWR cores that contain 8x8R fuel. This information is currently being reviewed by the staff for generic application. Although the evaluation is not yet complete, it was noted that the critical power test conditions specifically representative of normal conditions during second-cycle fuel operations exhibit a slightly non-conservative bias in predictions. This observation focuses in on a correlation behavioral concern not explicitly addressed in the overall GETAB methods approval (8) for the 7x7 and 8x8 fuel types. However, based on our review to date, we conclude that the 8x8R GEXL correlation has an acceptability and applicability equivalent to those of the 7x7 and 8x8 GEXL correlations previously approved by the staff.

Therefore, we conclude that there is sufficient conservatism implicit in the generic determination of the 1.07 safety limit MCPR to offset a possible nonconservatism associated with this concern for Cycle 4 of Hatch 1. Additionally, the generic evaluation [4] considered an all 8x8R equilibrium core, whereas the Cycle 4 core involves 7x7, 8x8 and 8x8R fuel in a non-equilibrium condition. In view of these conservatisms (which are representative of a typical non-equilibrium 8x8R reload core) we believe that the overall thermal-hydraulic (GETAB) methods are adequate for establishing conservative MCPR operating limits for Cycle 4.

2.2.2.2 Transient Analysis Results

The transients evaluated were the limiting pressure and power increase transient (turbine trip without bypass in this case), the limiting coolant temperature decrease transient (loss of a feedwater heater), the feedwater controller failure transient, and the control rod withdrawal error transient. Initial conditions and transient input parameters as specified in Sections 6 and 7 of Reference 3 were assumed.

The calculated system responses and ΔCPRs for the transients and conditions listed above have been analyzed by the licensee. Results were as follows:

	ΔCPR 7x7	ΔCPR 8x8	ΔCPR 8x8R
Turbine Trip Without Bypass	.06	.10	.10
Loss of 100°F Feedwater Heater	.13	.14	.14
Feedwater Controller Failure	.06	.07	.07
Rod Withdrawal Error	.19	.17	.12
Fuel Loading Error, Rotated Bundle*	<.09	<.09	.09

*The mislocated bundle error is considered separately in Section 2.3.3.

The above analyses include the effect of an End of Cycle recirculation pump trip (RPT) initiated by turbine stop valve closure or throttle valve fast closure. This RPT feature inserts negative reactivity into the reactor due to the rapid flow decrease and resultant increased voiding. Thus, the RPT helps shut down the reactor, effectively increasing the effectiveness of turbine-initiated scrams.

The transient analyses described above were performed with the REDY code.[9] A new improved code, ODYN,[16] has been developed by GE. The ODYN code, which uses a more physically correct model of the plant, generally predicts smaller ΔCPRs than the REDY code when the transient under study is fairly severe. However, as transient severity is lessened, ODYN predicts a greater ΔCPR than REDY (Reference 10, p. 1). Both codes are run with conservative input values, but ODYN is a better predictor of plant behavior once these input values are specified.[17]

GE has stated (Reference 10) that REDY can still be used because the limiting transient has a ΔCPR sufficiently large to be above the region where REDY is non-conservative with respect to ODYN. We have proceeded on this basis in approving reloads thus far.

The addition of the RPT feature has significantly reduced the ΔCPR associated with transients involving a turbine trip. (Reductions as great as roughly a factor of 2 are presented on p. 12 of Reference 10.) This improvement has brought the reload 3 transient analysis into the region where GE's assertion[10] is no longer valid for those transients which involve a turbine trip. The limited data available to us clearly indicate that calculations which include axial effects and detailed steam line modeling predict more severe results than do point kinetics REDY calculations.

Therefore, unless more justification for the REDY-based calculations is forthcoming, the transient analysis results must be conservatively adjusted to account for this effect. The analyses affected are the turbine trip without bypass transient and the feedwater controller failure transient. The loss of feedwater heater transient is much slower, and therefore should be well simulated by point kinetics calculations. Moreover, although the loss of a feedwater heater leads to a reactor trip in this case, there is no turbine trip and thus no significant excitation of acoustic resonances in the steam line. The remaining analyses (rod withdrawal error and rotated bundle) are not calculated with REDY and therefore are not affected.

Thus, the turbine trip transient and the feedwater controller failure transient (which involves a turbine trip) must have their analyses adjusted to account for defects in the steam line and core axial response modeling. Comparisons of the REDY and ODYN calculations presented on p. 12 of Reference 10 have enabled us to estimate a non-conservative trend for the REDY-calculated ΔCPR values. Accordingly, we will require that the ΔCPR values used in the calculation of the operating limit MCPR be adjusted upwards for those transients involving a turbine trip. This results in the following ΔCPRs:

	ΔCPR 7x7	ΔCPR 8x8	ΔCPR 8x8R
Turbine Trip Without Bypass	.11	.14	.14
Loss of 100°F Feedwater Heating	.13	.14	.14
Feedwater Controller Failure	.11	.12	.12
Rod Withdrawal Error	.19	.17	.12
Fuel Loading Error, Rotated Bundle	<.09	<.09	.09

Addition of the most severe ΔCPR to the safety limit (1.07) gives the appropriate operating limit MCPR for each fuel type. This will assure that the safety limit MCPR is not violated due to transients or fuel loading errors. Using the revised table of ΔCPRs, the operating limits become:

1.26 for 7x7 fuel (based on rod withdrawal error)

1.24 for 8x8 fuel (based on rod withdrawal error)

1.21 for 8x8R fuel (based on turbine trip without bypass and loss of 100°F feedwater heating)

These values are numerically equal to those originally proposed by the licensee, since the adjustments discussed above did not make the affected analyses limiting.

These values are based on predicted End-of-Cycle (EOC) reactor kinetics with prompt Recirculation Pump Trip (RPT). We have not completed our review of the hardware implementation of the prompt RPT feature for Hatch 1. Therefore, complete credit for the prompt RPT cannot be given.

The prompt RPT provides a reduction in recirculation flow which causes increased core void and thus introduces negative reactivity to mitigate transient power increases. It is most effective in this objective at EOC, since cycle-dependent core characteristics result in large void reactivity feedback towards EOC.

With regard for the effect of the prompt RPT on transient consequences, we have reviewed previous, applicable transient analyses (Reference 18) which was approved by Amendment No. 52. We have concluded that core characteristics, e.g., void coefficient and scram worth, which are important inputs to the transients for which the prompt RPT has been designed, conservatively bound the present core characteristics. The results of these analyses show that the transient consequences would be bounded by the current ΔCPR results until about 1000 MWd/t before EOC. Therefore, the operating limit MCPR's will be conservative up to 2000 MWd/t before EOC. The Technical Specifications have been amended to reflect this condition. MCPR limits for 2000 MWd/t-EOC4 to EOC 4 will be determined after completion of our review of the hardware implementation.

2.2.3.1 Thermal Power Monitor

The Thermal Power Monitor (TPM), also called an APRM Simulated Thermal Power (STP) Trip in some documents, is a modification to the APRM trip system. The modified system generates two trips: a trip with a flow biased setpoint and a second trip with a setpoint fixed at 120% power. The flow biased setpoint is unchanged from that presently in the Technical Specifications. However, the TPM conditions the APRM output to apply a time constant of about six seconds, which is less than but comparable to the fuel thermal time constants (seven to ten seconds). Thus, the signal compared to the flow biased setpoint is a conservative simulaton of fuel rod heat flux. This feature allows the plant operator to avoid spurious trips due to minor neutron flux overshoots when maneuvering the reactor.

The signal which is compared to the fixed 120% power setpoint is not modified. Thus, there is always a "fast scram" at 120% power in addition to the "heat flux" scram which may be below 120% power, depending on flow.

2.2.3.2 Effect of TPM on Safety Analyses

Since all the transient analyses are done assuming full design flow, the TPM has no effect because the 120% "fast" trip is identical to the original system at full power. Any effect due to the TPM must be on analyses which are initiated from low flow conditions.

GE has addressed the analysis of the various transients initiated from low flow conditions on pp. 5-8 of Reference 4. The generic analyses described there show that only the idle recirculation pump startup, recirculation flow controller failure (increasing), feedwater controller failure (max demand), and rod withdrawal error can become more severe at low flow conditions. This is the basis for the flow-dependent multiplier (K_f) in every GE plant's Technical Specifications.

The analyses supporting the K_f factor did not take credit for the flow biasing, but instead conservatively assumed the trip to occur at 120% power.[12] Therefore, the analyses supporting the flow-dependent multiplier (K_f) remain bounding.

Similarly, the various accident analyses which involve a neutron flux induced trip (e.g., rod drop accident) assume the trip to occur at 120% power regardless of initial power or flow conditions. Therefore, the validity of the accident analyses is also unaffected by the introduction of the TPM.

2.3 Accident Analyses

2.3.1 ECCS Appendix K Analysis

For the previous cycle, the licensee re-evaluated the adequacy of Hatch 1 ECCS performance in connection with the retrofit 8x8 reload fuel design.[13] The methods used in this analysis were previously approved by the staff. For that cycle, we reviewed the ECCS analysis results submitted by the licensee and concluded that Hatch 1 would be in conformance with all the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 when operated in accordance with the MAPLHGR 8x8R versus Average Planar Exposure values which appeared in the proposed plant Technical Specifications.[14] Since the Cycle 4 reload fuel is of the same design as that loaded for Cycle 3, we find this same LOCA-ECCS safety analysis and related Technical Specifications to be equally acceptable for showing compliance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 40 for the current Cycle 4 reload fuel.

2.3.2 Control Rod Drop Accident

For Hatch 1 Cycle 4, the accident reactivity shape function (cold) does not satisfy the requirements for the bounding analyses described in Reference 4. Therefore, it was necessary for the licensee to perform a plant and cycle specific analysis for the control rod drop accident. The results of this analysis indicated that the peak fuel enthalpy for this event would be at most 197.35 calories per gram.[3] Since this is well below the criterion of 280 calories per gram, we find the results of this analysis to be acceptable.

2.3.3 Fuel Loading Error

Potential fuel loading errors involving misoriented bundles have been explicitly included in the calculation of the operating limit MCPR. Potential errors involving bundles loaded into incorrect positions have also been analyzed by a method which considers the initial MCPR of each bundle in the core, and the resultant MCPR was shown to be greater than 1.07. This GE method for analysis of misoriented and misloaded bundles has been reviewed and approved by the staff.[15] The analyses which have been performed for potential fuel loading errors for Hatch 1 Cycle 4 are acceptable for assuring that CPRs will not be below the safety limit MCPR of 1.07.

2.3.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 4. As specified in the staff evaluation included in Reference 4, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow for the failure of at least one valve. Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable.

2.4 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis[3] show that the channel hydrodynamic and reactor core decay ratios at the natural circulation - 105% rod line intersection (which is the least stable physically attainable point of operation) are below the stability limit.

Because operation in the natural circulation mode at greater than 1% rated thermal power will be prohibited by Technical Specifications 3.6.J.1, there will be added margin to the stability limit and this is acceptable.

2.5 Startup Test Program

The licensee has not changed his startup test program from that approved for the previous cycle. This program therefore remains acceptable.

3.0 Environmental Consideration

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

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the 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: August 6, 1979

References

1. Letter, Chas. F. Whitmer (GPC) to Director of Nuclear Reactor Regulation (NRC), dated March 22, 1979.

2. "Proposed Changes to Technical Specifications," Attachments 1 and 2 of Reference 1.

3. "Supplemental Reload Licensing Submittal for Edwin I. Hatch Nuclear Plant Unit 1 Reload 3," NEDO-24175, January 1979.

4. "General Electric Boiling Water Reactor Generic Reload Application," NEDE-24011-P-A, May 1977.

5. Letter, R. E. Engel (GE) to U. S. Nuclear Regulatory Commission, dated January 30, 1979.

6. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, November 1973.

7. "Basis for 8x8 Retrofit Fuel Thermal Analysis Application," NEDE-24131, enclosed in letter, R. Gridley (GE) to D. Eisenhut and D. Ross (NRC), dated October 5, 1978.

8. Letter, W. Butler (NRC) to I. Stuart (GE), dated October 2, 1974.

9. "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, February 1973.

10. "Impact of One-Dimensional Transient Model on Plant Operations Limits," enclosure of letter, E. D. Fuller (GE) to U. S. Nuclear Regulatory Commission, dated June 26, 1978.

11. "Basis for Installation of Recirculation Pump Trip System," NEDO-24119, April 1978.

12. "Technical Specifications and Bases for Edwin I. Hatch Nuclear Plant Unit 1," Appendix A to Operating License DPR-47, Specification 2.1.A.1.C.

13. "Loss of Coolant Accident Report for Edwin I. Hatch Nuclear Plant Unit 1," NEDO-24086, December 1977, enclosure 2 of letter, C. Whitmer (GPC) to V. Stello (NRC), dated January 5, 1978.

14. "Edwin I. Hatch Nuclear Plant Unit No. 1 Reload 2 Safety Evaluation Report," April 11, 1978.

15. Letter, D. G. Eisenhut (NRC) to R. Engel (GE), dated May 8, 1978.

16. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, October 1978.

17. Letter, R. J. Mattson (NRC) to General Electric Company (Attn. G. G. Sherwood), dated March 20, 1979.

18. Charnly, J., "Supplemental Reload Licensing Submittal for Edwin I. Hatch Nuclear Plant Unit 1 Reload 2," NEDO-24078, Class 1, November 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

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. 69 to Facility Operating License No. DPR-57 issued to Georgia Power Company, Oglethorpe Electric Membership Corporation, Municipal Electric Association of Georgia, and City of Dalton, Georgia, which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 1, (the facility) located in Appling County, Georgia. The amendment is effective as of the date of issuance.

The amendment consists of changes to the Technical Specifications to (1) permit operation of the facility during Cycle 4 with 164 reload fuel assemblies of the GE 8X8 retrofit design, and (2) permit modification of the APRM trip system by incorporating a Thermal Power Monitor.

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

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

For further details with respect to this action, see (1) the application for amendment dated March 22, 1979, amended May 11, and 16, 1979, (2) Amendment No. 69 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 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 6th day of August 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors