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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. 31 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit 1. The amendment consists of changes to the Technical Specifications in response to your application dated February 23, 1976, as supplemented by letters dated March 8, 1976 and March 22, 1976, March 29, 1976.

This amendment consists of changes to the Technical Specifications associated with facility modifications which will improve the functioning of the Low Pressure Coolant Injection System.

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

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

Sincerely,

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

Enclosures:
1. Amendment No. 31
2. Safety Evaluation
3. Federal Register Notice

OFFICE➤	ORB#3	ORB#3	OELD	ORB#3		
SURNAME➤	CParrish	JGuibert:acr	C. Nissl	GLear		
DATE➤	3/25/76	3/25/76	3/30/76	3/30/76		

Georgia Power Company & Oglethorpe
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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 1

AMENDMENT TO FACILITY OPERATING LICENCE

Amendment No. 31
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 February 23, 1976, and supplemented by letters dated March 8, 1976 and March 22, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: March 30, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 31

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace pages 3.2-5, 3.2-8, 3.2-11, 3.2-12, 3.2-35, 3.2-59 through 3.2-62, 3.5-3, 3.6-10 and 3.6-22 with the attached revised pages. No change has been made on pages 3.2-7, 3.2-36 and 3.5-4.

Table 3.2-2

INSTRUMENTATION WHICH INITIATES OR CONTROLS HPCI

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1.	Reactor Water Level (Yarway)	Low Low (LL2)	2	$\geq$-38 inches	Initiates HPCI; also initiates RCIC
2.	Drywell Pressure	High	2	$\leq$2 psig	Initiates HPCI; also initiates LPCI and Core Spray and provid a permissive signal to ADS.
3.	HPCI Turbine Overspeed	Mechanical	1	$\leq$5000 rpm	Trips HPCI turbine
4.	HPCI Turbine Exhaust Pressure	High	1	$\leq$150 psig	Trips HPCI turbine
5.	HPCI Pump Suction Pressure	Low	1	$\leq$15" Hg vacuum	Trips HPCI turbine
6.	Reactor Water Level (Narrow Range)	High	2	$\leq$+58 inches	Trips HPCI turbine
7.	HPCI System Flow (Flow Switch)	High	1	$>$800 gpm	Closes HPCI minimum flow bypas line to suppres
		Low	1	$\leq$500 gpm	Opens HPCI minimum flow bypass line if pressure permissive is present.
8.	HPCI Equipment Room Temperature	High	1	$\leq$90^{o}F + ambient	Closes isolation valves in HPC system, trips HPCI turbine.
9.	HPCI Equipment Room Differential Temperature	High	1	$\leq$50^{o}F	Closes isolation valves in HPC system, trips HPCI turbine.

Notes for Table 3.2-2 (Cont.)

b. Whenever any CCCS subsystem is required to be operable by Section 3.5, there shall be two operable trip systems. If the required number of operable channels cannot be met for one of the trip systems, that system shall be repaired or the reactor shall be placed in the Cold Shutdown Condition within 24 hours after this trip systems is made or found to be inoperable.

Table 3.2-3

INSTRUMENTATION WHICH INITIATES OR CONTROLS RCIC

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1	Reactor Water Level (Yarway)	Low Low (LL2)	2	≥ -38 inches	Initiates RCIC: also initiates HPCI
2	RCIC Turbine Overspeed	Electrical	1	≤ 110% rated	Trips RCIC turbine
		Mechanical	1	≤ 125% rated	Trips RCIC turbine
3	RCIC Turbine Exhaust Pressure	High	1	≤ +25 psig	Trips RCIC turbine
4	RCIC Pump Suction Pressure	Low	1	≤ 15" Hg vacuum	Trips RCIC turbine
5	Reactor Water Level (Narrow Range)	High	2	≤ +58 inches	Trips RCIC turbine
6	RCIC System Flow (Flow Switch)	High	1	> 80 gpm	Closes RCIC minimum flow bypass line to suppression chamber
		Low	1	≤ 40 gpm	Opens RCIC minimum flow bypass line if pressure permissive is present
7	RCIC Equipment Room Temperature	High	1	≤ 90° F + ambient	Closes isolation valves in RCIC system, trips RCIC turbine
8	RCIC Equipment Room Differential Temperature	High	1	≤ 50° F	Closes isolation valves in RCIC system, trips RCIC turbine.
9	RCIC Steam Line Pressure	Low	2	_50 psig	Closes isolation valves in RCIC system, trips RCIC turbine.

Table 3.2-5

INSTRUMENTATION WHICH INITIATES OR CONTROLS THE LPCI MODE OF RHR

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1	Reactor Water Level (Yarway)				
		Low Low Low (LL3)	2	≥-146.5 inches	Initiates LPCI mode of RHR
2	Drywell	High	2	≤2 psig	Initiates LPCI mode of RHR
3	Reactor Pressure	High (Shutdown Cooling Mode)	1	≤135 psig	With primary containment isolati signal, closes RHR (LPCI) inboar motor operated injection valves
		Low	2	≤335 psig	Permissive to close Recirculatio Discharge Valve and Bypass Valve
		Low	2	≤500 psig	Permissive to open LPCI injectio valves
4	Reactor Water Level (Shroud Level Indicator)		1	>2/3 Core Height (313.5 inches)	Indicates _ 2/3 Core Height and acts as permissive to divert some LPCI flow to containment spray
5	LPCI Cross Connect Valve Open Annunciator	N/A	1	Valve not closed	Initiates annunciator when valve is not closed

Table 3.2-5 (Cont.)

INSTRUMENTATION WHICH INITIATES OR CONTROLS THE LPCI MODE OF RHR

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
6	RHR (LPCI) Pump Discharge Pressure Interlocks		2	$\geq$100 psig	Defer ADS actuation until LPCI or CS pump is confirmed to be running
7	RHR (LPCI) Pump Flow (ΔP Switch)	Low	1	$\geq$400 gpm	Opens LPCI minimum flow line upon receipt of low flow signal from both pumps and closes LPCI minimum flow line when signal from either pump is not present
8	RHR (LPCI) Pump Start Timers		1	$0 < t < 1$ seconds	With loss of normal power, and upon receipt of emergency power, one RHR pump starts immediately, the other three follow in 10 seconds
			1	$9 < t < 11$ seconds	
9	Valve Selection Timers				
			1	$\geq$ 10 minutes	Cancels LPCI injection valve initiation signal
10	RHR Relay Logic Power Failure Monitor		1	Not Applicable	Monitors availability of power to logic system

Table 4.2-5

Check, Functional Test, and Calibration Minimum Frequency for Instrumentation Which Initiates or Controls the LPCI Mode of RHR

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	Reactor Pressure	None	(d)	Every 3 months
4	Reactor Water Level (Shroud Level Indicator)	Once/day	(d)	Every 3 months
5	LPCI Cross Connect Valve Open Annunciator	None	Once/Operating cycle	None
6	RHR (LPCI) Pump Discharge Pressure Interlock	None	(d)	Every 3 months
7	RHR (LPCI) Pump Flow (Flow Switch)	None	(d)	Every 3 months
8	RHR (LPCI) Pump Start Timers	None	N/A	Once/operating cycle
9	Valve Selection Timers	None	N/A	Once/operating cycle
10	RHR Relay Logic Power Failure Monitor	None	Once/operating cycle	None

Notes for Table 4.2-5

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 4.2-5 and items in Table 3.2-5.

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. LPCI Subsystem
2. Containment Spray Subsystem

3.2.D. 3. RHR Pump Discharge Pressure High

An RHR pump discharge pressure of $\geq$ 100 psig indicates that LPCI flow is available when the reactor is depressurized. The presence of this signal means low pressure core standby cooling is available. Low pressure core standby cooling available is one of the four signals required to initiate ADS.

4. Core Spray Pump Discharge Pressure High

A core spray pump discharge pressure of $\geq$ 100 psig indicates that Core Spray flow is available when the reactor is depressurized. The presence of this signal means low pressure core standby coolin is available. Low pressure core standby cooling available is one of the four signals required to initiate ADS.

5. Auto Depressurization Timer

The 120-second delay time setting is chosen to be long enough so that the HPCI system has time to start, yet not so long that the core spray system and LPCI are unable to adequately cool the core if HPCI fails to start. An alarm in the main control room is annunciated each time either of the timers is timing. Resetting the automatic depressurization system logic in the presence of tripped initiating signals recycles the timers.

6. Automatic Blowdown Control Power Failure Monitor

The Automatic Blowdown Control 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.

E. Instrumentation Which Initiates or Controls the LPCI Mode of RHR (Table 3.2-5)

1. Reactor Water Level

b. Reactor Water Level Low Low Low (LL3) (Yarway)

Reactor vessel low water level (LL3) initiates LPCI and indicates that the core is in danger of being overheated because of an insufficient coolant inventory. This level is sufficient to allow the timed initiation of the various valve closure and loop selection routines to go to completion and still successfully perform its design function.

2. Drywell Pressure High

Primary containment high pressure is indicative of a break in the nuclear system process barrier inside the drywell. The high drywell pressure setpoint of $\leq$ 2 psig is selected to be high enough to avoid spurious starts but low enough to allow timely system initiation.

3. Reactor Pressure Low

The Bases for Reactor Pressure (Shutdown Cooling Mode) are discussed in the Bases for Specification 3.2.A.2.

With Recirculation Discharge Valve closure initiation delayed until reactor pressure has decayed to less than 335 psig the differential pressure across the closed valve will always be less than the maximum design 200 psid.

Once the LPCI system is initiated, a reactor low pressure setpoint of 500 psig produces a signal which is used as a permissive to open the LPCI in-

3.2.E.3. Reactor Pressure Low (Continued)

jection valves. The valves do not open, however, until reactor pressure falls below the discharge head of LPCI.

4. Reactor Water Level (Shroud Level Indicator)

A reactor water level $\geq$ 2/3 core height is indicative that LPCI has made progress in reflooding the core. A simultaneous high drywell pressure trip indicates the need for containment cooling. The $\geq$ 2/3 core height setpoint acts as a permissive for manual diversion for some of the LPCI flow to containment spray.

5. LPCI Cross Connect Valve Open Annunciator

With the modified LPCI arrangement, the cross connect valve status was changed from normally open to normally closed. Inadvertent opening of this valve could negate the LPCI system injection when needed. The annunciator will alarm when the LPCI cross connect valve is not fully closed.

6. RHR (LPCI) Pump Discharge Pressure Interlocks

A pressure $\geq$ 100 psig on the RHR 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. RHR (LPCI) Pump Flow (Δp Switch) Low

A flow switch is provided downsteam of each RHR 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 for each pair of pumps. A single moter-operated valve controls the condition of each bypass line. The minimum flow bypass valve automatically opens upon sensing low flow in the discharge lines from both pumps of the associated pump pair. The valve automatically closes whenever the flow from either of the associated main system pumps is above the low flow setting.

8. RHR (LPCI) Pump Start Timers

If normal AC power is available, four pumps automatically start without delay. If normal AC power is not available, one pump starts without delay as soon as power becomes available from the standby sources. The other three pumps start after a 10-second delay. The timer provides correct sequencing of the loads to the diesel generator.

3.2.E.9. Valve Selection Timers

After 10 minutes, a timer cancels the LPCI signals to the injection valves. The cancellation of the signals allows the operator to divert the water for other post-accident purposes. Cancellation of the signals does not cause the injection valves to move.

10. RHR Relay Logic Power Failure Monitor

The RHR Relay 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.

F. Instrumentation Which Initiates or Controls Core Spray (Table 3.2-6)

1. Reactor Water Level Low Low Low (LL3) (Yarway)

A reactor low water level of -146.5 inches on the Yarway (LL3) initiates Core Spray. This level is indicative that the core is in danger of being overheated because of an insufficient coolant inventory.

2. Dyrwell Pressure High

Primary containment high pressure is indicative of a break in the nuclear system process barrier inside the drywell. The high drywell pressure setpoint of $\leq$ 2 psig is selected to be high enough to avoid spurious system initiation but low enough to allow timely system initiation.

3. Reactor Pressure Low

Once the core spray system is initiated, a reactor low pressure setpoint of 500 psig produces a signal which is used as a permissive to open the core spray injection valves. The valves do not open, however, until reactor pressure falls below the discharge head of the core spray system.

4. Core Spray Sparger Differential Pressure

A detection system is provided to continuously confirm the integrity of the core spray piping between the inside of the reactor vessel and the core shroud. A differential pressure switch measures the pressure difference between the top of the core support plate and the inside of the core spray sparger pipe just outside the reactor vessel. If the core spray sparger piping is sound, this pressure difference will be the pressure drop across the core resulting from inter-channel leakage. If integrity is lost, this pressure drop will include the steam separator pressure drop. An increase in the normal pressure drop initiates an alarm in the main control room.

3.5.B.1. Normal System Availability (Continued)

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

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

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

2. Operation with Inoperable Components

a. One Pump Inoperable

If one RHR pump is inoperable, the reactor may remain in operation for a period not to exceed thirty (30) days provided the remaining RHR pumps, access paths of the RHR system, the CS system, and the diesel generators are operable.

4.5.B.1. Normal Operational Tests

	Item	Frequency
b.	Simulated Automatic Actuation Test	Once/Operating Cycle
c.	System flow rate: Three RHR pumps shall deliver at least 23,000 gps against a system head of at least 20 psig.	Once/3 months
d.	Pump Operability	Once/month
e.	Motor Operated valve operability	Once/month

2. Surveillance with Inoperable Components

a. One Pump Inoperable

When one RHR pump is inoperable, the remaining RHR pumps and active components in access paths of the diesel generators shall be demonstrated to be operable immediately. The operable RHR system pumps shall be demonstrated to be operable daily thereafter until the inoperable pump is returned to normal service.

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

b. Two Pumps Inoperable

If two RHR pumps are inoperable, the reactor may remain in operation for a period not to exceed seven (7) days provided the CS system, all active components of both RHR subsystems, and the associated diesel generators are operable.

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

b. Two Pumps Inoperable

When two pumps are inoperable the CS system, all active components of both RHR subsystems and containment cooling subsystems, and diesel generators shall be demonstrated to be operable immediately and daily thereafter until the inoperable subsystem is returned to normal service.

4.6.I. Jet Pumps (Continued)

2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

3. The diffuser to lower plenum differential pressure reading on an individual jet pump vary from the mean of all jet pump differential pressures by more than 10%.

3.6.J. Recirculation Pump Speeds

1. Operation with a single recirculation pump is permitted for 24 hours unless the recirculation pump is sooner made operable. If the pump cannot be made operable, the reactor shall be in cold shutdown within 24 hours.

2. Following one pump operation the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

K. Structural Integrity of Primary System Boundary

The structural integrity of the primary system boundary shall be maintained at the level required to assure safe operation throughout the life of the unit. The reactor shall be maintained in a Cold Shutdown Condition until each indication of a defect has been investigated and evaluated.

J. Recirculation Pump Speeds

Recirculation pump speeds shall be recorded at least once per day.

K. Structural Integrity of Primary System Boundary

A preservice inspection of accessible components listed in Table 4.6-1 will be conducted before initial fuel loading to establish a preservice base for later inspections. The nondestructive inspections listed in Table 4.6-1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

3.5.I Jet Pumps (Continued)

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY AND OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-321

Introduction

By letter dated February 23, 1976, and by supplemental letters dated March 8 and March 22, 1976, Georgia Power Company (GPC) requested an amendment to Facility Operating License No. DPR-57 for Edwin I. Hatch Nuclear Plant Unit 1 which would revise the Technical Specifications to reflect proposed modifications to the Low Pressure Coolant Injection (LPCI) system.

Discussion

The proposed modifications to the LPCI system involve:

1. Selection Logic and Cross-tie Valve

   The removal of the recirculation loop selection logic, closing the cross-tie valve between the two LPCI system discharge piping headers, locking open the associated valve motor circuit breaker, and providing an annunciator to indicate an open condition of the cross-tie valve;

2. Discharge Valve Closure

   Changing the action of the recirculation loop discharge valves and discharge bypass valves such that they all are required to close upon signal to the modified LPCI system;

3. Injection Signals and Valve Operation

   Rewiring of the system so that the automatic initiation signals direct both of the normally closed injection valves to open upon detection of LOCA conditions;

4. Power Supply

   Separating the electrical power supply system for the LPCI system valve motors into two motor control centers, one for recirculation loop "A" valves and the other for recirculation loop "B" valves; and

5. LPCI Pump Power Supply

Changing the power supplies for LPCI pumps "C" and "D" such that LPCI pump "C" will be assigned as an electrical load in Division 1 and LPCI pump "D" will be assigned to Division II electrical loads.

These modifications are designed to increase the reliability and availability of the LPCI system in the event of a postulated loss-of-coolant accident (LOCA), thereby improving the overall performance of the integrated Emergency Core Cooling System. The NRC staff has reviewed and approved similar modifications at Vermont Yankee, Brunswick Unit 2, Fitzpatrick, and Peach Bottom Units 2 and 3.

Evaluation

The NRC staff's evaluation of the proposed changes to the LPCI system are as follows:

1. Selection Logic and Cross-tie Valve

The loop selection logic circuitry of the LPCI system will be removed from the control room panels. Removal of this logic circuitry allows both LPCI injection valves to open, given an accident signal, no matter where the recirculation pipe break is located. This situation of opening both injection valves requires that the RHR cross-tie valve remain closed during normal plant operations and accident conditions. The application has proposed that (a) the keylock switch on the control room panel which operates the cross-tie valve will be changed from "keylock open" to keylock close", (b) the cross-tie valve circuit breaker at the motor control center cubical will be padlocked open with the valve closed and, (c) an annunciator will be added to alarm whenever the cross-tie valve is open. We find these proposed changes to be an acceptable method of assuring that this valve will remain closed during normal plant operation or accident conditions and are, therefore, acceptable.

2. Discharge Valve Closure

Changing the function of the recirculation loop discharge valves such that they are both required to close at the time of the postulated LOCA was reviewed by the NRC staff to determine the capability of the valve to operate under the accident conditions. For the modified LPCIS to perform its function as designed, the NRC staff required assurance that the discharge valves would not receive the permissive signal to closure until the reactor vessel pressure had decayed to less than 365 psig and would not experience a differential pressure of greater than 200 psig during closure. The information submitted by the licensee for Hatch Unit 1 has satisfied the NRC staff that a differential pressure greater than 200 psig will not occur.

3. Injection Signals and Valve Operation

Due to the elimination of the loop selection logic, the accident initiation signals have been rewired to direct (1) both LPCI injection valves to open, and (2) both recirculation loop discharge valves and discharge bypass valves to close when reactor pressure decreases to an appropriate setting.

The LPCI system redundant injection valves, pumps and recirculation valves are controlled by ac control power relays in their control circuitry. These relays are in turn controlled by redundant 125-volt dc output relays provided in each actuation train in the LPCI logic panels. This assures that failure of the 120-volt dc power supply of either train will not prevent operation of any valve and pump in either train. Separation has been provided within the logic panels and wiring between the two logic panels is run through separate conduit. Separation of A & B circuits is maintained by the conduit so that any assumed failure of a conduit run will not prevent the operation of the redundant or associated control systems. We conclude that these design changes do not compromise the separation and independence of the two safety trains and are acceptable.

4. Power Supply

a. Use of Motor Control Centers

In the existing onsite (standby) power system for Hatch Unit 1, the motor operators for both LPCI system injection valves and for both recirculation loop discharge valves and discharge bypass valves are powered from swing bus MCC S018 which has swing capabilities between 600 volt electrical busses 1C and 1D which are connected to diesel generators 1A and 1C respectively. As a part of the proposed modifications, swing bus MCC S018 will be separated into two motor control centers (MCC S018A and MCC S018B). MCC S018A will supply power to loop "A" LPCI valves and loop "A" recirculation loop discharge and discharge bypass valves; MCC S018B will supply power to loop "B" LPCI valves and loop "B" recirculation loop discharge and discharge bypass valves. During the postulated LOCA condition, MCC S018A would be supplied by diesel generator 1A through 600 volt bus 1C and MCC S018B would be supplied by diesel generator 1C through 600 volt bus 1D. Backup power supply to either MCC S018A or MCC S018B is provided through a swing bus arrangement from diesel generator 1B.

Separation of the LPCI system valve motor operators, as discussed above, into two motor control centers improves the capability of the LPCI system to sustain a single electrical failure without loss of design function and is acceptable.

b. Swing Bus Transfer, Surveillance and License Commitment

Our review of the swing bus arrangement associated with the backup power to the LPCI system injection valves has shown it not to be in conformance with Regulatory Guide 1.6, "Independence Between Standby Power Sources and Their Distribution Systems". We have informed the licensee that this design is questionable in view of the present application for an improved LPCI system. The improved LPCI system could allow Hatch I to obtain credit for additional reflood capability following the Design Basis LOCA. The original emergency power distribution system design was reviewed and approved by the NRC staff based upon the fact that no credit would be given for the LPCI system contribution to the core reflood capability following a LOCA (i.e., with the original recirculation loop selection logic in effect, the single failure of a LPCI injection valve in conjunction with a postulated LOCA could prevent the LPCI system from injecting water into the core). Therefore, the NRC staff now considers it appropriate to require the elimination of the swing bus arrangement as an additional safety feature on those facilities applying for the LPCI modification.

We further informed the licensee that separate, independent and redundant power sources for the LPCI injection valves would be required to bring the 600 volt portion of the onsite emergency power system into conformance with the recommendations of Regulatory Guide 1.6. However, we will allow the licensee a reasonable period of time to institute their modified design after our review and approval.

We consider operation of Hatch I with the presently proposed system acceptable in the interim because:

(1) In order to alleviate the problem of a potential undetected failure in the swing bus transfer circuitry that could prevent the bus from transferring to its alternate source, we have placed a requirement in the technical specifications for surveillance on these transfer circuits: the transfer circuits will be tested for operability once every two months.

(2) In order to minimize the potential for an inadvertent paralleling of two diesel generators through a swing bus, GPC has provided redundant interlocks in the opening and closing mechanisms of the circuit breakers which effect the swing bus power transfer.

By letter dated March 29, 1976, the licensee has agreed to make the necessary modifications to his present design so that the modified design meets Regulatory Guide 1.6.

This commitment from the licensee, in conjunction with the additional design features of the swing bus transfer arrangement and the technical specification surveillance requirements for the transfer circuits, represents an acceptable short-term solution to the electrical power supply system problem and justifies continued operation of Hatch 1 in the interim. We will review the design changes proposed to meet Regulatory Guide 1.6 when they are submitted.

5. LPCI Pump Power Supply

The wiring changes associated with the modifications to the power supplies for LPCI pumps "C" and "D" will be accomplished in such a manner as to assure that required separation and independence of the two safety train divisions are provided.

6. LPCI Pump Flow Tests

As a result of the LPCI modifications, the operating modes of the LPCI pumps will be changed such that two pumps discharge into each injection header thereby changing the discharge flow characteristics from that previously established. Prior to reactor startup the licensee will conduct flow tests to establish pump discharge path characteristics from which pump flow curves will be developed. This information will be used to assure satisfaction of pump net positive suction head requirements.

7. Operating Limits and Continued Reactor Operation

GPC has proposed to operate Hatch Unit 1, following the completion of the LPCI system modifications, in accordance with the current operating limits which were established in conjunction with a previously approved ECCS evaluation (License Amendment No. 27, December 17, 1975). We have reviewed the ECCS capability of Hatch Unit 1 with the modified LPCI system, assuming the occurrence of the most limiting single failure of ECCS equipment. We have determined that the availability of the integrated core cooling systems is improved with the modified LPCI system and is equal to or better than that assumed in the ECCS evaluation approved on December 17, 1975.

Consequently, since the currently existing operating limits conservatively bound the Hatch Unit 1 ECCS performance with the modified LPCI system, we have concluded that continued reactor operation in accordance with these current limits, following the completion of the LPCI system modifications, is acceptable.

8. Environmental Aspects

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 statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 30, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY

OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 31 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 1, located in Appling County, Georgia. The amendment is effective as of its date of issuance.

The amendment consists of changes to the Technical Specifications associated with facility modifications which will improve the functioning of the Low Pressure Coolant Injection System.

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

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

For further details with respect to this action, see (1) the application for amendment dated February 23, 1976, and supplemented by letters dated March 8, 1976 and March 22, 1976, (2) Amendment No. 31 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 30 day of March, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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