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OCT 29 1975

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. 21 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit 1. The amendment also incorporates Change No. 21 in the Technical Specifications in accordance with your applications dated April 29, 1975 and August 13, 1975.

This amendment will (1) revise the surveillance testing requirements for the personnel air lock to the primary containment and (2) correct a typographical error.

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

Sincerely,

/s/

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

Enclosures:
1. Amendment No. 21
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

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SURNAME ➤	CParrish	JGuibert:esp		GLear		
DATE ➤	10/22/75	10/22/75	10/24/75	10/29/75		

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Georgia Power Company &
Oglethorpe Electric Membership Corporation

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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 LICENSE

Amendment No. 21
License No. DPR-57

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

   A. The applications for amendments by Georgia Power Company and Oglethorpe Electric Membership Corporation (the licensees) dated April 29, 1975 and August 13, 1975, comply 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.

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

"(1) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 21."

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 Reactor Licensing

Attachment:
Change No. 21 to the
Technical Specifications

Date of Issuance: OCT 2 9 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 21

CHANGE NO. 21 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace page 1.1-15, page 1.2-1, page 3.5-9, page 3.5-18, page 3.6-21 and page 3.7-27 with the attached revised pages. No change has been made on page 1.1-16, page 3.5-10, page 3.6-22 and page 3.7-28.

2.1.A.6. Main Steam Line Isolation Valve Closure on Low Pressure (Continued)

Advantage is taken of the scram trip feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity Safety Limit. Operation of the reactor at pressures lower than 880 psig requires that the reactor Mode Switch be in the START & HOT STANDBY position, where protection of the fuel cladding integrity Safety Limit is provided by IRM's and the APRM 15% scram (Start and Hot Standby Mode.) Thus, the combination of main steam line low pressure isolation and isolation valve closure scram trip assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity Safety Limit. The reactor pressure vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the Run Mode is less severe than the loss of feedwater analyzed in section 14.3 of the FSAR, therefore closure of the MSIV's for thermal transient protection when not in the Run Mode is not required.

7. Main Steam Line Isolation Valve Closure on Low Condenser Vacuum

To provide backup protection for the main condenser against overpressure due to in-leakage, assuming that the turbine stop valves and bypass valves fail to close, a loss of condenser vacuum initiates automatic closure of all main steam isolation valves, the main steam line drain isolation valve and the reactor water sample line valve (i.e. initiates a Group 1 isolation). Closure of these valves prevents excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system. The low vacuum trip set point is selected far enough above the normal operating vacuum to avoid spurious isolation, however, low enough to provide backup isolation prior to the rupture of the condenser.

This trip function is active in any reactor mode. The main steam line isolation valve closure on low condenser vacuum trip is inactive only when all of the following conditions exist: (1) the low condenser vacuum switch is in BYPASS; (2) the turbine stop valves are less than 90% open; and (3) reactor pressure is less than 1045 psig.

B. Reactor Water Level Trip Settings Which Initiate Core Standby Cooling Systems (CSCS)

The core standby cooling systems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each Core Standby Cooling System component was established based on the reactor water low level scram trip setting. To lower the trip setting of the water low level scram trip would increase the capacity requirement for each of the CSCS components. Thus, the reactor vessel water low level scram trip was set low enough to permit margin for operation, yet will not be set lower because of CSCS capacity requirements.

The design of the CSCS components to meet the above guidelines was dependent upon three previously set parameters: the maximum break size, the water low level scram trip setting and the CSCS initiation trip setting. To lower the trip setting for initiation of the CSCS may lead to a decrease in effective core cooling. To raise the CSCS initiation trip setting would be in a safe direction, but it would reduce the margin established to prevent actuation of the CSCS during normal operation or during normally expected transients. Transient and accident analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margin for the fuel.

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

The Safety Limit, established to preserve the reactor coolant system integrity, applies to the limit on the reactor vessel steam dome pressure.

Objective

The objective of the Safety Limit (associated with preserving the reactor coolant system integrity) is to establish a pressure limit below which the integrity of the reactor coolant system is not threatened due to any overpressure condition.

Specifications

A. Reactor Vessel Steam Dome Pressure

1. When Irradiated Fuel is in the Reactor

   The reactor vessel steam dome pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the reactor vessel steam dome pressure Safety Limit 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 reactor vessel steam dome pressure Safety Limit from being exceeded.

Specifications

A. Nuclear System Pressure

1. When Irradiated Fuel is in the Reactor

   When irradiated fuel is present in the reactor vessel, and the head is bolted to the vessel, the limiting safety system settings shall be as specified below:

Protective Action		Limiting Safety System Settings (psig)
a. Scram on high reactor pressure (reactor vessel steam dome pressure)		$\leq$ 1045
b. Nuclear system relief valves open on nuclear system pressure	4 valves	$\leq$1091
	4 valves	$\leq$1101
	3 valves	$\leq$1111

21

3.5.F Automatic Depressurization System (ADS)

1. Normal System Availability

Six of the seven valves of the Automatic Depressurization System shall be operable:

a. Prior to reactor startup from a cold shutdown, or

b. When there is irradiated fuel in the reactor vessel and the reactor is above 113 psig except as stated in Specification 3.5.F.2.

2. Operation with Inoperable Components

If two of the seven ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed thirty (30) days, provided the HPCI system is operable. (Note that the pressure releif function of these valves is assured by Specification 3.6.H; Specification 3.5.F only applies to the ADS function). 21

3. Shutdown Requirements

If Specification 3.5.F.1 or 3.5.F.2 cannot be met, an orderly shutdown will be initiated and the reactor pressure shall be reduced to 113 psig or less within 24 hours.

4.5.F Automatic Depressurization System (ADS)

1. Normal Operational Tests

A simulated automatic actuation test shall be performed on the ADS prior to startup after each refueling outage. Surveillance of all relief valves is covered in Specification 4.6.H.

2. Surveillance with Inoperable Components

When it is determined that two of the seven ADS valves are incapable of automatic operation, the HPCI system and the actuation logic of the other ADS valves shall be demonstrated to be operable immediately and every seven (7) days thereafter until six ADS valves are capable of automatic operation.

3.5 Minimum Core and Containment Cooling Systems Availability

During any period when one of the standby diesel generators is inoperable, continued reactor operation is limited to seven (7) days unless operability of the diesel generator is restored within this period. During such seven (7) days all of the components in the RHR system LPCI mode and containment cooling mode shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours. Specification 3.9 provides further guidance on electrical system availability.

Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown Condition, both core spray systems and the LPCI and containment cooling subsystems of the RHR system may be inoperable provided that the shutdown cooling subsystem of the RHR system is operable in accordance with Specification 3.5.B.1.b and that no work is being done which has the potential for draining the reactor vessel.

H. Maintenance of Filled Discharge Pipes

Whenever the core spray system, LPCI, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. The suction of the HPCI pumps shall be aligned to the condensate storage tank.

4.5.G Surveillance of Core and Containment Cooling Systems

When it is determined that one of the standby diesel generators is inoperable, the remaining diesels and all of the components in the RHR system LPCI mode and containment cooling mode connected to the operable diesel generators shall be demonstrated to be operable immediately and daily thereafter.

H. Maintenance of Filled Discharge Pipes

The following surveillance requirements shall be performed to assure that the discharge piping of the core spray system, LPCI, HPCI, and RCIC are filled when required:

1. Every month prior to the testing of the LPCI and core spray systems, the discharge piping of these systems shall be vented.

F.1. Normal System Availability (continued)

Specification 3.6.H states the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation facilures yet be fully capable of performing their pressure relief function. 21

Because the automatic depressurization system does not provide makeup to the reactor primary vessel, no credit is taken for the system cooling of the core caused by the system actuation to provide further conservatism to the Core Standby Cooling Systems. Performing analysis of the automatic depressurization system is considered only with respect to its depressurizing effect in conjunction with LPCI or core spray and is based on five valves. There are seven valves connected to the ADS circuitry. Since credit was taken for only five in the performance evaluation of the ADS, it is appropriate that one valve may be out indefinitely without appreciably lowering the probability that the system will perform satisfactorily.

2. Operation with Inoperable Components

With two ADS valves known to be incapable of automatic operation (only one of these may be failed in a way that negates the pressure relief function as stated ub Specification 3.6.H) five valves remain operable to perform their ADS function. However, because of the difficulty in proving the operability of the ADS function (actuation of the ADS for testing would cause an unnecessary system blowdown). Reactor operation is only allowed to continue for thirty (30) days providing the HPCI is demonstrated to be operable and the actuation logic of the five ADS remaining valves is demonstrated to be operable. 21

3. Minimum Core and Containment Cooling Systems Availability

The purpose of this Specification is to assure that adequate core cooling equipment is available at all times. If, for example, one core spray loop were out of service and the diesel which powered the opposite core spray were out of service, only 2 RHR pumps would be available. Specification 3.9 must also be consulted to determine other requirements for the diesel generators. In addition, refer to definition 1.0.00 for Cumulative Downtime requirements.

This specification establishes conditions for the performance of major maintenance, such as draining of the suppression pool. The availability of the shutdown cooling subsystem of the RHR system and the RHR service water system ensure adequate supplies of reactor cooling and emergency makeup water when the reactor is in the Cold Shutdown condition. In addition this specification provides that, should major maintenance be performed, no work will be performed which could lead to draining the water from the reactor vessel.

3.6.H. Relief/Safety Valves (Continued)

Experience in relief/safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failure or deteriorations. The relief/safety valves are benchtested every second operating cycle to ensure that their set points are within the tolerance given in Specification 2.2.A.1.b. The relief/safety valves are tested in place at low reactor pressure once per operating cycle to establish that they will open and pass steam. 21

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

I. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within $\pm$ 5%, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

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.

J. Recirculation Pump Speeds

The LPCI loop selection logic has been previously described in FSAR, Section 7.4.3.5. For some limited low probability accidents with the recirculation loo½ operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

Analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. Below 80% power the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

Notes For Tables 3.7-2 through 3.7-4

(1) Test duration for all valves and penetrations listed will generally exceed one hour.

(2) Test pressures are at least 59 psig for all valves and penetrations except MSIV's which are tested at 28 psig.

(3) MSIV acceptable leakage limit is 11.5 scfh/valve of air.

(4) The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.6 L_a.

(5) Local leak tests on all testable isolation valves shall be performed each major refueling shutdown but in no case at intervals greater than 2 years.

(6) Local leak tests on all testable penetrations shall be performed each major refueling shutdown but in no case at intervals greater than 2 years.

(7) The personnel air lock shall be tested at intervals not to exceed 6 months.

(8) The personnel air lock door seals are tested at 10 psig after each opening. 21

(9) Identifies isolation valves that are tested by applying pressure between the inboard and outboard isolation valves. Inboard valve is not tested in the direction required for isolation but will have equivalent or more conservative leakage results.

(10) Identifies isolation valves that are tested by applying pressure between the isolation valve and a manually operated valve such that the isolation valve is tested in the direction required for isolation.

(11) Identifies isolation valves that are tested by applying pressure between the isolation valves and other system valves. Isolation valves not tested in the direction required for isolation will have equivalent or more conservative results.

3.7. CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the emergency core cooling systems in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the initial startup test program is being conduted. There will be no pressure on the system at this time, which greatly reduces the changes of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occurring. Procedures for rod withdrawal and patterns programmed into the Rod Worth Minimizer and Rod Sequence Control System would limit control rod worth to less than 1.25% Δk. A drop of such a rod does not result in any fuel damage. In addition in the unlikely event that an excursion did occur, the secondary containment (reactor building) and standby gas treatment system, which shall be operational during this time, offers a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

1. Pressure Suppression Chamber

The pressure suppression chamber water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdowns.

Since all of the non-condensible gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compressure plus the vapor pressure of the liquid must not exceed 62 psig, the maximum pressure. The design volume of the pressure suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. Reference FSAR Section 5.2.3.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 TO FACILITY OPERATING LICENSE NO. DPR-57

(CHANGE NO. 21 TO TECHNICAL SPECIFICATIONS)

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

DOCKET NO. 50-321

Introduction

By letters dated April 29 and August 13, 1975, Georgia Power Company (GPC) requested changes to the Technical Specifications appended to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit 1. The proposed changes include:

1. A revision to the notes for Table 3.7-2, "Testable Penetrations with Double O-Ring Seals", which reflects a change to the surveillance testing requirements for the personnel air lock to the primary containment which was previously approved in License Amendment No. 9.

2. The correction of a typographical error in Technical Specification 3.5.F.2, "Automatic Depressurization System - Operation with Inoperable Components".

In addition, GPC has requested that the bases for the Technical Specifications be revised to:

a. Reflect the change in the safety settings for the nuclear coolant system relief valves which was approved in License Amendment No. 8.

b. More accurately describe the conditions for activation and deactivation of the main steam line isolation valve trip on low condenser vacuum.

c. Correct a typographical error in the bases for Technical Specification 3.5.F.2.

Evaluation

Our evaluation of the proposed changes is as follows:

1. Notes for Table 3.7-2

   The NRC staff had previously approved, by License Amendment No. 9, a revision to the surveillance requirements (section 4.7.A.2.e) for testing the personnel air lock to the primary reactor containment. That revision increased the emphasis on testing the air lock door seals rather than on testing the air lock itself. A change to the notes for Table 3.7-2 which describe the testing requirements for the air lock was inadvertently not included in License Amendment No. 9. The proposed change would administratively update the Technical Specifications to include the information which was omitted. Since this proposal reflects a previously approved revision to the surveillance requirements, we conclude that it is acceptable.

2. Automatic Depressurization System (ADS)

   Technical Specification 3.5.F.2, Limiting Condition for Operation with Inoperable ADS Valves, notes that the pressure relief function of these same valves (dual function safety/relief valves) is assured by Specification 3.6.D. However, Specification 3.6.D refers to Idle Recirculation Loop Startup whereas Specification 3.6.H refers to the subject Safety/Relief Valves. The proposed change would correct this typographical error to ensure that the proper Technical Specification (i.e., 3.6.H) is referenced. Since this proposal corrects an existing error in the Technical Specifications, we conclude that it is acceptable.

3. Changes to the Bases

   We have reviewed the proposed changes and have concluded that they are acceptable in that they improve the clarity and accuracy of the bases.

Conclusion

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

Date: OCT 2 9 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 21 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 will (1) revise the surveillance testing requirements for the personnel air lock to the primary containment and (2) correct a typographical error.

The applications for the amendment comply 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.

For further details with respect to this action, see (1) the applications for amendment dated April 29, 1975 and August 13, 1975, (2) Amendment No. 21 to License No. DPR-57, with Change No. 21, 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 Reactor Licensing.

Dated at Bethesda, Maryland, this 27th day of October, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

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SURNAME➤						
DATE➤						

For further details with respect to this action, see (1) the applications for amendment dated April 29, 1975 and August 13, 1975, (2) Amendment No. 21 to License No. DPR-57, with Change No. 21, 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 Reactor Licensing.

Dated at Bethesda, Maryland, this 29th day of October, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

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

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DATE➤	10/20/75	10/22/75	10/24/75	10/29/75		