MAY 24 1978

Distribution
Docket
ORB #3
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ACRS (16)
OPA (CMiles)
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JRBuchanan

Docket No. 50-321

Georgia Power Company
Oglethorpe Electric Membership Corporation
Municipal Electric Association of Georgia
City of Dalton, Georgia
ATTN: Mr. Charles F. Whitmer
Vice President-Engineering
Georgia Power Company
Atlanta, Georgia 30302

Gentlemen:

In response to your request for license amendment dated November 1, 1976 as amended by letter dated April 15, 1977, the Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit 1.

This amendment incorporates provisions into the facility Technical Specifications which establish limiting conditions for operation and surveillance requirements for drywell to suppression chamber differential pressure control.

These requirements provide assurance that facility operation will be in accordance with the assumptions utilized in your facility's plant-unique analysis which was performed in conjunction with the Mark I Containment Short Term Program evaluation.

The enclosed license amendment reflects those changes to your original request for license amendment which have been agreed to in discussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities.

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

Sincerely,

Original signed by

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

Concurrence per DD memo dtd 4/21/78

Const. 1
GD

Enclosures and ccs:
See page 2

OFFICE	ORB #3	ORB #3	DOR	OELD	ORB #3
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DATE	5/23/78	5/ /78	5/23/78	5/ /78	5/24/78

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Enclosures:
1. Amendment No. 55
2. Safety Evaluation
3. Notice

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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. 55
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 November 1, 1976, as amended by letter dated April 15, 1977, 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. 55, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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: May 24, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 55

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	Replace
3.2-22	3.2-22
3.2-23	3.2-23
3.2-48	3.2-48
3.7-10	3.7-10
------	3.7-10a
------	3.7-33a
3.7-34	3.7-34

Amendment No. 42, 46, 55

Table 3.2-11

INSTRUMENTATION WHICH PROVIDES SURVEILLANCE INFORMATION

Ref. No. (a)	Instrument (b)	Required Operable Instrument Channels	Type and Range	Action	Remarks
1	Reactor Water Level (GE/MAC)	1 2	Recorder Indicator 0 to 60"	(c) (c)	(d) (d)
2	Shroud Water Level	1 1	Recorder Indicator +200" to +500"	(c) (c)	(d) (d)
3	Reactor Pressure	1 2	Recorder Indicator 0 to 1200 psig	(c) (c)	(d) (d)
4	Drywell Pressure	2	Recorder -5 to +80 psig	(c)	(d)
5	Drywell Temperature	2	Recorder 0 to 500°F	(c)	(d)
6	Suppression Chamber Air Temperature	2	Recorder 0 to 500°F	(c)	(d)
7	Suppression Chamber Water Temperature	2	Recorder 0 to 250°F	(c)	(d)
8	Suppression Chamber Water Level	2 2	Indicator 0 to 300" Recorder 0 to 30"	(c) (c)(e)	(d) (d)
9	Suppression Chamber Pressure	2	Recorder -5 to +80 psig	(c)	(d)
10	Rod Position Information System (RPIS)	1	28 Volt Indicating Lights	(c)	(d)
11	Hydrogen and Oxygen Analyzer	1	Recorder 0 to 5%	(c)	(d)
12	Post LOCA Radiation Monitoring System	1	Recorder Indicator 1 to 10^6 R/hr	(c) (c)	(d) (d)
13	Drywell/Suppression Chamber Differential Pressure	2	Recorder -0.5 to +2.3 psid	(c)(e)	(d)

NOTES FOR TABLE 3.2-11

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-11 and items in Table 4.2-11.

b. Limiting Conditions for Operation for the Neutron Monitoring System are listed in Table 3.2-7.

c. From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

Continued operation is permissible for seven days from and after the date that one of these parameters is not indicated in the control room. Surveillance of local panels will be substituted for indication in the control room during the seven days.

d. Drywell and Suppression Chamber Pressure are each recorded on the same recorders. Each output channel has its own recorder.

Drywell and Suppression Chamber air temperature and suppression chamber water temperature are all recorded on the same recorders. Each output channel has its own recorder. Each recorder takes input from several temperature elements.

Hydrogen and Oxygen are indicated on one recorder. The recorder has two pens, one pen for each parameter.

Each channel of the post LOCA radiation monitoring system includes two detectors; one located in the drywell and the other in the suppression chamber. Each detector feeds a signal to a separate log count rate meter. The meter output goes to a two pen recorder. One high radiation level alarm is provided per channel and annuciation of alarm is provided in the control room.

e. In the event that all indications of this parameter is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

Table 4.2-11

Check and Calibration Minimum Frequency for Instrumentation
Which Provides Surveillance Information

Ref. No. (a)	Instrument	Instrument Check Minimum Frequency (b)	Instrument Calibration Minimum Frequency (c)
1	Reactor Water Level (GE/MAC)	Each shift	Every 6 months
2	Shroud Water Level	Each shift	Every 6 months
3	Reactor Pressure	Each shift	Every 6 months
4	Drywell Pressure	Each shift	Every 6 months
5	Drywell Temperature	Each shift	Every 6 months
6	Suppression Chamber Air Temperature	Each shift	Every 6 months
7	Suppression Chamber Water Temperature	Each shift	Every 6 months
8	Suppression Chamber Water Level	Each shift	Every 6 months
9	Suppression Chamber Pressure	Each shift	Every 6 months
10	Rod Position Information System (RPIS)	Each shift	N/A
11	Hydrogen and Oxygen Analyzer	Each shift	Every 6 months
12	Post LOCA Radiation	Each shift	Every 6 months
13	Drywell/Suppression Chamber Differential Pressure	Each shift	Every 6 months

3.7.A.6.c. H_2 and O_2 Analyzer

Whenever the reactor is in power operation, there shall be at least one CAD System H_2 and O_2 analyzer serving the primary containment. If one H_2 and O_2 analyzer is inoperable, the reactor may remain in operation for a period not to exceed seven days.

d. Post-LOCA Repressurization Limit

The maximum post-LOCA primary containment repressurization limit allowable using the CAD System shall be 30 psig. Venting via the SGTS to the main stack must be initiated at 30 psig following the initial post-LOCA pressure peak.

7. Drywell-Suppression Chamber Differential Pressure

Differential pressure between the drywell and suppression chamber shall be maintained equal to or greater than 1.5 psid except as specified in (1) and (2) below: If this specification cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

1) This differential pressure shall be established within 24 hours after having placed the Mode Switch in the RUN mode. The differential pressure may be removed within 24 hours prior to achieving a shutdown.

2) This differential pressure may be decreased to less than 1.5 psid for a maximum of four hours during required operability testing of the HPCI system pump, the RCIC system pump, and the drywell-pressure suppression chamber vacuum breakers.

4.7.A.6.c. H_2 and O_2 Analyzer

Instrumentation Surveillance is listed in Table 4.2-11.

7. Drywell-Suppression Chamber Differential Pressure

The pressure differential between the drywell and suppression chamber shall be recorded once each shift.

8. Shutdown Requirements

If Specification 3.7.A cannot be met, an orderly shutdown shall be initiated and the reactor shall be brought to Hot Shutdown within 12 hours and shall be in the Cold Shutdown condition within the following 24 hours.

3.7.B. Standby Gas Treatment System

1. Operability Requirements

Two independent standby gas treatment systems shall be operable at all times when secondary containment integrity is required.

After one of the standby gas treatment systems is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system shall be demonstrated to be operable within 2 hours and daily thereafter.

If the system is not made fully operable within 7 days, reactor shutdown shall be in cold shutdown within the next 36 hours and fuel handling operations shall be terminated within 2 hours.

4.7.B. Standby Gas Treatment System

1. Surveillance When System Operable

At least once per operating cycle, not to exceed 18 months, the following conditions shall be demonstrated:

a. Pressure drop across the combined HEPA filters and charcoal absorber banks is less than 6 inches of water at the system design flow rate (+10%, -0%).

b. Operability of inlet heater at rated power when tested in accordance with ANSI N510-1975.

c. Air distribution is uniform within 20% across the filter train when tested in accordance with ANSI N510-1975.

3.7.A.7. Drywell-Suppression Chamber Differential Pressure

The Drywell to Torus Differential Pressure (DTDP) System is designed to maintain a differential pressure between the drywell and the suppression chamber during normal operation in order to reduce the water leg in the suppression chamber downcomers. The need for this differential pressure was identified by the plant unique analysis (PUA) which was performed for the HNP-1 suppression chamber in order to evaluate the potential effects of Post-LOCA hydrodynamic loads. The reduction of the water leg in the suppression chamber downcomers caused by the differential pressure was found to be one method for limiting the Post-LOCA hydrodynamic loads to a value less than the structural acceptance criteria identified by the PUA.

The DTDP system consists of a gas compression circuit located outside of the primary containment which takes suction on the suppression chamber and discharges to the drywell. The circuit contains redundant 100% capacity compressors and associated valves and controls to ensure plant availability.

Maintaining the required differential pressure identified in the PUA ensures that Post-LOCA loads will not exceed the structural acceptance criteria identified in the PUA.

3.7.A.8. Shutdown Requirements

Bases for shutdown requirements are discussed above in conjunction with the individual requirements for primary containment integrity.

B. Standby Gas Treatment System

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. Both standby gas treatment system fans are designed to automatically start upon a high radiation signal from either the refueling floor ventilation exhaust duct monitor or the reactor building ventilation exhaust duct monitor or upon receipt of a signal from the primary containment isolation system. In addition, the system can also be manually started from the main control room. Upon receipt of any of the isolation signals, both fans start, all systems isolation valves open and each fan draws air from the isolated reactor building. One train is manually placed in the standby mode after at least one train is operable. The standby train restarts automatically upon loss of flow in the operating train. One fan will maintain the differential pressure and all leakage past the secondary containment will be inleakage. Each of the two fans has 100% capacity. A detailed discussion of the standby gas treatment system may be found in Section 5.3.3.3 of the FSAR.

Only one of the two standby gas treatment systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant should be placed in a condition that does not require a standby gas treatment system.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accident analyzed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 55 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

I. INTRODUCTION

In conjunction with the Short Term Program (STP) evaluation of Boiling Water Reactor facilities with the Mark I containment system, the Georgia Power Company (the licensee) submitted a Plant Unique Analysis (PUA) for the Edwin I. Hatch Nuclear Plant Unit No. 1. This analysis was performed to confirm the structural and functional capability of the containment suppression chamber and attached piping, to withstand newly-identified suppression pool hydrodynamic loading conditions which had not been explicitly considered in the original design analysis for the plant. As part of the STP evaluation, specific loading conditions were developed for each Mark I facility, to account for the change in the magnitude of the loads due to plant-specific variations from the reference plant design for which the basic loading conditions were developed.

The results of the staff's review of the hydrodynamic load definition techniques and the Mark I containment plant unique analyses are described in the "Mark I Containment Short Term Program Safety Evaluation Report," NUREG-0408, December 1977. As discussed in this report, the NRC staff has concluded that each Mark I containment system would maintain its integrity and functional capability in the unlikely event of a design basis loss-of-coolant accident (LOCA) and, therefore, that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately two years, while a methodical, comprehensive Long Term Program is conducted.

As discussed in Section III.C of NUREG-0408, of all of the plant parameters that were considered in the development of the hydrodynamic loads for the STP, only two parameters are expected to vary during normal plant operation; these are (1) the drywell-wetwell differential pressure; and (2) the suppression chamber (torus) water level. Subsequent to the submittal of the PUA, the licensee was requested to submit proposed Technical Specifications which assure that the allowable range of these two parameters during facility operation would be in accordance with the values utilized in the PUA.

The licensee has been operating this facility with differential pressure control to enhance the safety margins of the containment structure since early 1976. This evaluation provides a more detailed basis for establishing the allowable range of drywell-wetwell differential pressure and torus water level, in order to quantify containment safety margins. This amendment incorporates these parameters into the Technical Specifications with the associated limiting conditions for operation and surveillance requirements.

By letter dated November 1, 1976, the licensee proposed changes to the facility Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for differential pressure control. The existing Technical Specifications include limiting conditions for operation on torus water level which are applicable to the results of the PUA. Our evaluation of these proposed changes follows.

II. EVALUATION

The licensee has proposed certain Technical Specification requirements for the purpose of assuring that the normal plant operating conditions are within the envelope of conditions considered in their PUA. These Technical Specification changes establish (1) limiting condition for operation (LCOs) for drywell to torus differential pressure, and (2) associated surveillance requirements. All other initial conditions utilized in the PUA are either presently included in the Technical Specifications or are configurational conditions which have been confirmed by the licensee and will not change during normal operation.

Differential pressure between the drywell and the suppression chamber will result in leakage of the drywell atmosphere to the lower pressure regions of the reactor building and to the torus airspace. This leakage from the drywell will cause a slow decay in the differential pressure. Therefore, surveillance requirements for the differential pressure have been included in the Technical Specifications. Surveillance frequency of once per operating shift for the differential pressure was selected on the basis of previous operating experience.

The torus water level is not expected to vary significantly during normal operation, unless certain systems connected to the suppression pool are activated. The torus water level would normally be monitored whenever such systems are in use. Therefore, we find that inclusion of periodic torus water level surveillance requirements in the Technical Specifications is not required.

We have reviewed the differential pressure and torus water level monitoring instrumentation systems proposed by the licensee with regard to the number of available channels and the instrumentation accuracy. This type of instrumentation is typically calibrated at six-month intervals. To assure proper operation during such intervals, two monitoring channels for both differential pressure and torus water level have been provided, such that a comparison of the readings will indicate when one of the channels is inoperative or drifting. The errors in the instrumentation are sufficiently small relative to the magnitude of the measurement (i.e., a maximum differential pressure measurement error of 0.1 psid in a measurement of 1.0 to 2.0 psid and a maximum torus water level measurement error of 10% of the difference between the maximum and minimum torus water level) that they may be neglected, based on the expected load variation with differential pressure and torus water level.

There are certain periods during normal plant operations when the differential pressure control cannot be maintained. Therefore, provisions have been included in the Technical Specifications to relax the differential pressure/control requirements during specified periods. The justification for relaxing the differential pressure control during these specific periods and the basis for selecting the duration of the periods are discussed in detail below.

A. Startup and Shutdown

During plant startup and shutdown, the drywell atmosphere undergoes significant barometric changes due to the variation in heat loads from the primary and auxiliary systems. In addition, it is during these periods that the drywell is being either inerted with nitrogen gas or deinerted. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have limited the relaxation of the differential pressure control requirements for the startup and shutdown periods to 24 hours following startup and 24 hours prior to a shutdown. This time period was selected on a basis similar to that for the inerting requirements, already existing in the Technical Specifications. The postulated design basis accident for the containment assumes that the primary system is at operating pressure and temperature. During the startup and shutdown transients, the primary system is at operating pressure and temperature for only part of the transient, during which the differential pressure is being established. These time periods have been shown by previous operating experience to be adequate with respect to the startup and shutdown transients, and at the same time sufficiently small in comparison to the duration of the average power run. Since the principal accident event to which differential pressure control is important to assure containment integrity (i.e., with a factor of safety of two) is a large break LOCA, we have considered whether there is a significantly greater probability of a large break LOCA during the startup and shutdown transients. We have concluded that there is not. Further, the operation of the plant systems is monitored more closely than normal during these periods and a finite magnitude of differential pressure will be available during the majority of these periods to mitigate the potential consequences of an accident.

B. Testing and Maintenance

During normal operation, there are a number of tests which are required to be conducted to demonstrate the continued functional performance of engineered safety features. The testing of certain systems will require, or result in, a reduction in the drywell-torus differential pressure. The operability testing of the drywell-torus vacuum breakers requires the removal of the differential pressure to permit the vacuum breakers to open. For the testing of high-energy systems (e.g. high pressure coolant injection pumps) during normal operation, the discharge flow is routed to the suppression pool. This energy deposition will raise the temperature of the suppression pool, resulting in an increase in torus pressure and a reduction in the differential pressure.

Functional performance testing of engineered safety features is necessary to assure proper maintenance of these systems throughout the life of the plant. Some of these tests (i.e., pump operability and drywell-wetwell vacuum breakers) may require or result in a reduction in the differential pressure. We estimate that not more than four tests will be required each month which will result in a reduction in differential pressure. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have permitted a relaxation of differential pressure control in order to conduct these tests, limited to a period of up to four hours. Again, we have carefully considered whether the probability of a large LOCA is significantly greater during these testing periods than that during normal operation. We conclude that it is not. Moreover, only the test of the drywell-wetwell vacuum breakers requires complete removal of the differential pressure.

Provisions have also been included in the Technical Secifications for performing maintenance activities on the differential pressure control system and for resolving operational difficulties which may result in an inadvertent reduction in the differential pressure for a short period of time. In certain circumstances, corrective action can be taken without having to attain a cold shutdown condition. To avoid repeated and unnecessary partial cooldown cycles, a restoration period has been incorporated into the action requirements of the LCO for differential pressure control; i.e., in the event that the differential pressure cannot be restored in six hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours. The six hour restoration period was selected on the basis that it represents an adequate minimum period of time during which any short-term malfunctions could be corrected, coupled with the minimum period of time required to conduct a controlled shutdown. The allowable time to conduct a controlled shutdown has been minimized, because the containment transient response is more a function of the primary system pressure than the reactor power level. On this basis, we find the proposed restoration period and action requirement acceptable.

We conclude that the limits imposed on the periods of time during which operation is permitted without the differential pressure control fully effective provides adequate assurance of overall containment integrity, and the periods of time differential pressure control is completely removed are acceptably small.

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 Section 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 the amendment.

CONCLUSIONS

The proposed Technical Specifications will provide the necessary assurance that the plant's operating conditions remain within the envelope of the conditions assumed in the Plant Unique Analysis (PUA) performed in conjunction with the Mark I Containment Short Term Program. The PUA supplements the facility's Final Safety Analysis Report (FSAR) in that it demonstrates the plant's capability to withstand the suppression pool hydraulic loads which were not explicitly considered in the FSAR. We therefore conclude that the proposed changes to the Technical Specifications are acceptable.

We further conclude, 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 amendment 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 activites 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: May 24, 1978

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. 55 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, located in Appling County, Georgia. The amendment is effective as of its date of issuance.

The amendment revised the Technical Specifications to incorporate requirements for establishing and maintaining the drywell to suppression chamber differential pressure to maintain the margins of safety established in the NRC staff's "Mark I Containment Short Term Program Safety Evaluation," NUREG-0408. Operation in accordance with the conditions specified in NUREG-0408 has been previously authorized in 43 F.R. 13108, March 29, 1978.

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) application for amendment dated November 1, 1976, as supplemented April 15, 1977, (2) Amendment No. 55 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 single 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 24th day of May 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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