OCT 1 6 1975

Docket No. 50-321

Georgia Power Company & Oglethorpe Electric
Membership Corporation
ATTN: Mr. I. S. Mitchell, III
Vice President & Secretary
Georgia Power Company
Atlanta, Georgia 30302

Gentlemen:

The Commission has issued the enclosed Amendment No. 16 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit 1. The amendment also incorporates Change No. 16 in the Technical Specifications in accordance with your application dated March 31, 1975 which was submitted in reply to our letter dated February 15, 1975. During our review of your response, a few changes were discussed and found mutually acceptable to you and to the NRC staff.

The amendment defines new temperature limits for the suppression pool water to provide additional assurance of maintaining primary containment integrity.

A copy of the related Federal Register Notice is also enclosed. A copy of the Safety Evaluation on this matter was transmitted to you with our letter dated July 16, 1975.

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 16
2. Federal Register Notice

cc: See next page

SEE PREVIOUS YELLOW FOR CONCURRENCE CHAIN

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The amendment defines new temperature limits for the suppression pool water to provide additional assurance of maintaining primary containment integrity.

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

Sincerely,

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

Enclosures:
1. Amendment No. 16
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

Subject to attached note
Done 10/6/75

OFFICE ➤	RL:ORB-3	RL:ORB-3	OELD	RL:ORB-3	RL:AD/ORs KRG	RL:DIR KRG/for
x7872 SURNAME ➤	CParrish	JGuibert:esp	S H Lewis	GLear GL	KRGoller	RSBoyd
DATE ➤	9/23/75 for	9/23/75	10/2/75	10/6/75	10/7/75	10/7/75

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Georgia Power Company &
Oglethorpe Electric Membership Corporation

OCT 16 1975

cc: w/enclosures

G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
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Georgia Power Company
Edwin I. Hatch Plant
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Mr. G. Wyman Lamb, Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

Mr. John Robins
Office of Planning and Budget - Room 615-C
270 Washington, Street, S. W.
Atlanta, Georgia 30334

Mr. Dave Hopkins
U. S. Environmental Protection Agency
Region 1V Office
1421 Peachtree Street, N. E.
Atlanta, Georgia 30309

Mrs. Fleets Taylor, Librarian
Appling County Public Library
Parker Street
Baxley, Georgia 31513

October 3, 1975

Note to: John C. Guibert, Operating
Reactors Branch No. 3
Division of Reactor Licensing

RE: HATCH 1 SUPPRESSION POOL WATER TEMPERATURE TECH SPEC CHANGE

I have reviewed this package simultaneously with my review of the same Tech Spec change on Monticello, Quad Cities, Cooper, Pilgrim, and Dresden. The second paragraph of the Federal Register Notice for those actions contains a more comprehensive description of the action being taken and I suggest you use that language. Subject to resolution of the above matter, I have no objections to the issuance of this amendment.

Steve
Stephen H. Lewis
Office of the Executive
Legal Director

10/6 Change made.
JCGuibert

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. 16
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 March 31, 1975, 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.

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 licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 17."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller /for

Roger S. Boyd, Acting Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 16 to the
Technical Specifications

Date of Issuance: OCT 1 6 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 16

CHANGE NO. 16 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Delete pages 3.7-1, 3.7-2 and 3.7-29 from the Appendix A Technical Specifications and insert the revised pages. (No change made on page 3.7-2).

Add page 3.7-1a.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability

The Limiting Conditions for Operation associated with containment systems apply to the operating status of the primary and secondary containment systems.

Objective

The objective of the Limiting Conditions for Operation is to assure the integrity of the primary and secondary containment systems.

Specifications

A. Primary Containment

1. Pressure Suppression Chamber

At any time that irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression chamber water level and water temperature shall be maintained within the following limits except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mwt.

a. Minimum water level - 12 feet, 2 inches.

b. Maximum water level - 12 feet, 6 inches.

c. During normal power operation, the suppression chamber water temperature shall be maintained $\lesssim 95^{o}F$. If this temperature limit is exceeded, pool cooling shall be initiated immediately.

16 If the water temperature cannot be restored to $\lesssim 95^{o}F$ within 24 hours, the reactor shall be shut down using normal shutdown procedures.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability

The Surveillance Requirements associated with containment systems apply to the primary and secondary containment integrity.

Objective

The objective of the Surveillance Requirements is to verify the integrity of the primary and secondary containment.

Specifications

A. Primary Containment

1. Pressure Suppression Chamber

a. The pressure suppression chamber water level, water temperature and air temperature shall be measured and recorded daily.

b. The interior painted surfaces above the level one foot below the normal water line of the pressure suppression chamber shall be visually inspected once per operating cycle. In addition, the external surfaces of the pressure suppression chamber shall be visually inspected on a routine basis for evidence of corrosion or leakage.

c. Whenever there is indication that a significant amount of heat is being added to the pressure suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated. 16

LIMITING CONDITIONS FOR OPERATION

d. During relief valve operation or testing of RCIC, HPCI, or other testing which adds heat to the suppression pool, the maximum water temperature shall not exceed 105°F. In connection with such testing, the pool temperature must be reduced within 24 hours to less than or equal to 95°F.

e. The reactor shall be scrammed
16 from any operating condition when the suppression pool temperature reaches 110°F. Operation shall not be resumed until the pool temperature is reduced to below the normal power operation limit specified in c. above.

f. During reactor isolation conditions the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

SURVEILLANCE REQUIREMENTS

d. Whenever there is indication that there was relief valve operation with the temperature of the suppression pool exceeding 160°F and the reactor primary coolant system pressure greater 16
than 200 psig, an external visual examination of the pressure suppression chamber shall be conducted before resuming power operation.

2. Primary Containment Integrity

Primary containment integrity is required:

a. Prior to withdrawing control rods for the purpose of going critical.

b. Whenever the reactor is critical.

c. Whenever the reactor water temperature is above 212°F and fuel is in the reactor vessel.

An exception is made while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt, during which time primary containment integrity is not required.

4.7.A.2. Leak Testing to Verify Primary Containment Integrity

Primary containment integrity shall be demonstrated by the following test procedures:

a. Type A Tests - Integrated Leak Rate Test (ILRT) *

Primary containment integrity is confirmed if the leak rate does not exceed the maximum allowable leak rate, L_a, of 1.2 weight percent of the contained air per 24 hours at the peak test pressure.

(1) Type A tests shall be performed under the program established in Appendix J of 10 CFR Part 50. (Reference 1).

* L_a - Maximum allowable peak pressure test leak rate - 1.2 weight percent per day
L_t - Maximum allowable reduced pressure test leak rate
L_{am} - Measured peak pressure test leak rate - values are subject to change with each ILRT performed
L_{tm} - Measured reduced pressure test leak rate - values are subject to change with each ILRT performed
L_{ao} - Allowable operational leak rate for peak pressure tests - values are subject to change with each ILRT performed
L_{to} - Allowable operational leak rate for reduced pressure tests - values are subject to change with each ILRT performed
(All leakage rates measured in weight percent of contained air per 24 hours)
P_a - Peak test pressure - 59 psig
P_t - Reduced test pressure - 29.5 psig

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is less than 59 psig which is below the maximum pressure of 62 psig. The minimum water level of 12 feet, 2 inches corresponds to a water volume of 87,300 cubic feet and a downcomer submergence of 3 feet, 8-1/2 inches. The maximum water level of 12 feet, 6 inches corresponds to a water volume of 90,380 cubic feet. The corresponding downcomer submergence is 4 feet 1/2 inch. Since the majority of the Bodega tests (reference 1) were run with a submergence of 4 feet and with complete condensation, this specification is adequate with respect to downcomer submergence.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool
water, operating procedures define that action to be taken in the event
a relief valve inadvertently opens or sticks open. As a minimum this
action shall include: (1) use of all available means to close the valve,
(2) initiate suppression pool water cooling heat exchangers, (3)
16 initiate reactor shutdown, and (4) if other relief valves are used to
depressurize the reactor, their discharge shall be separated from that
of the stuck-open relief valve to assure mixing and uniformity of energy
insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 16 TO FAÇILITY OPERATING LICENSE NO. DPR-57

(CHANGE NO. 16 TO THE TECHNICAL SPECIFICATIONS)

GEORGIA POWER COMPANY AND

OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-321

Introduction

By letter dated March 31, 1975, Georgia Power Company (the licensee) requested a change to the Technical Specifications appended to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit 1. The proposed change was submitted in response to our request to the licensee, by letter dated February 15, 1975, for a proposed change to the Technical Specifications associated with pressure suppression pool water temperature limits. Certain modifications to the proposed change were made with mutual concurrence between the licensee and the NRC staff, to improve its clarity and intent.

Discussion

The Edwin I. Hatch Nuclear Plant Unit 1 is a boiling water reactor (BWR) which is housed in a Mark I primary containment. The Mark I primary containment is a pressure suppression type of primary containment that consists of a dry well and a suppression chamber (also referred to as the torus). The suppression chamber, or torus, contains a pool of water and is designed to suppress the pressure during a postulated loss-of-coolant (LOCA) by condensing the steam released from the reactor primary system. The reactor system energy released by relief valve operation during operating transients also is released into the pool of water in the torus.

Experiences at various BWR plants with Mark I containments have shown that damage to the torus structure can occur from two phenomena associated with relief valve operations. Damage can result from the forces exerted on the structure when, on first opening the relief valves, steam and the air within the vent are discharged into the torus water. This phenomenon is referred to as steam vent clearing. The second source of potential structural damage stems from the vibrations which accompany extended relief valve discharge into the torus water if the pool water is at elevated temperatures. This effect is known as the steam quenching vibration phenomenon.

1. <u>Steam Vent Clearing Phenomenon</u>

   With regard to the steam vent clearing phenomenon, we are actively reviewing this generic problem and in our letter dated February 15, 1975, we also requested each applicable licensee to provide information to demonstrate that the torus structure will maintain its integrity throughout the anticipated life of the facility. Because of apparent slow progression of the material fatigue associated with the steam vent clearing phenomenon, we have concluded that there is no immediate potential hazard resulting from this type of phenomenon; nevertheless, surveillance and review action on this matter by the NRC staff will continue in due course during this year.

2. <u>Steam Quenching Vibration Phenomenon</u>

   The steam quenching vibration phenomenon became a concern as a result of occurrences at two European reactors. With torus pool water temperatures increased in excess of 170F due to prolonged steam quenching from relief valve operation, hydrodynamic fluid vibrations occurred with subsequent moderate to high relief valve flow rates. These fluid vibrations produced large dynamic loads in the torus structure and extensive damage to torus internal structures. If allowed to continue, the dynamic loads could have resulted in structural damage to the torus itself, due to material fatigue. Thus, the reported occurrences of the steam quenching vibration phenomenon at the two European reactors indicate that actual or incipient failure of the torus can occur from such an event. Such failure would be expected to involve cracking of the torus wall and loss of containment integrity. Moreover, if a LOCA occurred simultaneously with or after such an event, the consequences could be excessive radiological doses to the public. In comparison with the steam vent clearing phenomenon, the potential risk associated with the steam quenching vibration phenomenon (1) reflects the fact that a generally smaller safety margin[1/] exists between the present license requirements on suppression pool temperature limits and the point at which damage could begin and (2) is more immediate.

1/ The difference, in pool water temperature, between the license limit(s) and the temperature at which structural damage might occur is the safety margin available to protect against the effects of the phenomenon discussed.

Evaluation

The existing Technical Specifications for Hatch Unit 1 limit the torus pool temperature to 95F. This temperature limit assures that the pool water has the capability to perform as a constantly available heat-sink with a reasonable operating temperature that can be maintained by use of heat exchangers whose secondary cooling water (the service cooling water) is expected to remain well below 95F. While this 95F limit provides normal operating flexibility, short-term temperatures permitted by operating procedures exceed the normal power operating temperature limit, but accommodates the heat release resulting from abnormal operation, such as relief valve malfunction, while still maintaining the required heat-sink (absorption) capacity of the pool water needed for the postulated LOCA conditions. However, in view of the potential risk associated with the steam quenching vibration phenomenon, it is necessary to modify the temperature limits now in the license Technical Specifications.

This action was, as discussed in our February 15, 1975 letter, first suggested by the General Electric Company (GE) who had earlier informed us of the steam quenching vibration occurrences at a meeting on November 1, 1974 and provided related information by letters to us dated November 7, and December 20, 1974. The December 20 letter stated that GE had informed all of its customers with operating BWR facilities and Mark I containments of the phenomenon and included in those communications GE's recommended interim operating temperature limits and proposed operating procedures to minimize the probability of encountering the damaging regime of the steam quenching vibration phenomenon.

Implementation of the GE recommended procedures and temperature limits by the proposed change to the Technical Specifications has been evaluated by the NRC staff as follows:

a. The new short-term limit applicable to all conditions requires that the reactor be scrammed if the torus pool water temperature reaches 110F. This new limit and associated requirement to scram the reactor provides additional margin below the 170F temperature related to potential damage to the torus.

b. For specific requirements associated with surveillance testing, i.e., testing of relief valves, the water temperature shall not exceed 10F above the normal power operation limit. This new limit applicable to surveillance testing of relief valves and RCIC or HPCI operation provides additional operating flexibility while still maintaining a maximum heat-sink capacity. The current limits in the Technical Specifications is a maximum suppression pool water temperature of 120F.

c. For reactor isolation conditions, the new temperature limit is 120F, above which temperature the reactor vessel is to be depressurized. This new limit of 120F assures pool capacity for absorption of heat released to the torus while avoiding undesirable reactor vessel cooldown transients. Upon reaching 120F, the reactor is placed in the cold, shutdown condition at the fastest rate consistent with the technical specifications on reactor pressure vessel cooldown rates.

d. In addition to the new limits on temperature of the torus pool water, discussion in the Bases includes a summary of operator actions to be taken in the event of a relief valve malfunction. These operating actions are taken in order to avoid the development of temperatures approaching the 170F threshhold for potential damage by the steam quenching phenomenon.

Conclusion

We have concluded, based on the consideration discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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 1 6 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. 16 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 incorporates additional suppression pool water temperature limits: (1) during any testing which adds heat to the pool, (2) at which reactor scram is to be initiated and (3) requiring reactor pressure vessel depressurization. It also adds surveillance requirements for visual examination of the suppression chamber during each refueling and following operations in which the pool temperatures exceed 160°F and add monitoring requirements of water temperatures during operations which add heat to the pool.

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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on July ?4, 1975 (40 F.R. 31045). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this action, see (1) the application for amendment dated March 31, 1975, (2) Amendment No. 16 to License No. DPR-57, with Change No. 16 and (3) the Commission's related Safety Evaluation issued on July 16, 1975. 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, Park Street, Baxley, Georgia.

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 10th day of October, 1975

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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