December 8, 1975

Docket No. 50-321

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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 requested the Federal Register to publish the enclosed Notice of Proposed Issuance of Amendment to Facility License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit 1. The proposed amendment includes a change to the Technical Specifications based on our letter to you dated September 23, 1975.

This amendment would revise the Technical Specifications to add requirements that would limit the period of time operation can be continued with immovable control rods that could have control rod drive mechanism collet housing failures.

A copy of our proposed license amendment with proposed changes to the Technical Specifications also is enclosed. A copy of our Safety Evaluation relating to this proposed action was forwarded to you with our letter dated September 23, 1975.

Sincerely,

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

Enclosures:
1. Federal Register Notice
2. Proposed Amendment w/Proposed Technical Specification changes

OFFICE	ORB#3	ORB#3	ORB#3	~~RL:AD/ORS~~		
SURNAME	CParrish	JGuibert:acr	GLear	~~KRGoller~~		
DATE	12/ 3 /75	12/ 4 /75	12/4 /75	~~12/ /75~~		

NUCLEAR REGULATORY COMMISSION

W'SHINGTON, D. C. 20555

GEORGIA POWER COMPANY AND

OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

PROPOSED AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.
License No. DPR-57

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

   A. 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

   B. 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:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendicies 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. ."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

Attachment:
Change No. to the
Technical Specifications
Date of Issuance: December 8, 1975

ATTACHMENT TO PROPOSED LICENSE AMENDMENT

PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Add page 3.3-1a. Replace pages 3.3-2 and 3.3-9 with new pages. No changes have been made on page 3.3-1 and page 3.3-10.

3.3 REACTIVITY CONTROL

Applicability

The Limiting Conditions for Operation associated with reactivity control apply to the operational status of the control rod system.

Objective

The objective of the Limiting Conditions for Operation is to assure the ability of the control rod system to control reactivity.

Specifications

A. Core Reactivity Margin

A sufficient number of control rods capable of insertion shall always be available so that when fully inserted, the core is subcritical for any reactivity condition during the operating cycle, with the highest worth control rod capable of withdrawal fully withdrawn.

B. Inoperable Control Rods

1. No Movement by Control Rod Drive Pressure

Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. Control rod drives falling within this category shall have their directional control valves disarmed electrically and the associated rod positions shall be accounted for in complying with specification 3.3.A.

4.3 REACTIVITY CONTROL

Applicability

The Surveillance Requirements associated with reactivity control apply to the control rod system.

Objective

The objective of the Surveillance Requirements is to verify the ability of the control rod system to control reactivity.

Specifications

A. Core Reactivity Margin

A sufficient number of analytically selected control rods shall be withdrawn following initial fuel loading or any refueling outage when core alterations are made to demonstrate with a margin of 0.38% Δk that the core can be made subcritical for any reactivity condition during the subsequent operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted.

B. Operable Control Rod Exercise Requirements

Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 30% rated thermal power. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day when operating above 30% rated thermal power.

3.3 REACTIVITY CONTROL

B. Inoperable Control Rods (Cont'd)

1. No Movement by Control Rod Drive Pressure (Cont'd)

If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to the Cold Shutdown condition within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing, and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.B.

If investigation demonstrates that the cause of control rod drive failure is a cracked collet housing or if that possibility cannot be eliminated, the reactor shall not be started until the affected control rod drive has been replaced or repaired.

3.3.B.2 Excessive Scram Time

Control rods with a scram insertion time for 90% insertion which exceeds 7.00 seconds shall be considered inoperable, but if they can be moved with control rod drive pressure, they need not be fully inserted or disarmed electrically.

3.3.B.3 Inoperable Accumulators

Control Rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable

4. Limiting Number of Inoperable Control Rods

During reactor power operation, no more than one control rod in any 5 x 5 array may be inoperable (at least 4 operable control rods must separate any 2 inoperable ones). If this Specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a shutdown condition within 24 hours.

C. Control Rod Drive System

1. Control Rod Drive Coupling Integrity

Each control rod shall be coupled to its drive or completely inserted and its directional control valves disarmed electrically except during control rod drive maintenance as stated in Specification 3.10.E.

4.3.B Operable Control Rod Exercise Requirements (Cont'd)

When it is initially determined that a control rod is incapable of normal insertion, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted the reactor shall be brought to the Cold Shutdown Condition within 24 hours and a shutdown margin test made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted.

Once per week, check the status of the pressure and level alarms for each accumulator.

4.3.C. Control Rod Drive System

1. Control Rod Drive Coupling Integrity

The coupling integrity shall be verified for each withdrawn control rod as follows:

a. When the rod is withdrawn the first time after each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation and rod position indication including where applicable the "full-in" and "full-out" position. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 30% power shall be performed to verify instrumentation response.

B. Control Rods

Limiting Conditions for Operation:

Specification 3.3.B.1 requires that a rod which cannot be moved with drive pressure be taken out of service by being disarmed electrically. To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication. If the rod is fully inserted and disarmed electrically, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A. This assures that the core can be shutdown at all times with the remaining control rods assuming the highest worth operable control rod does not insert. An allowable pattern for control rods disarmed electrically, which shall meet this Specification, will be determined and made available to the operator. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housing, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing will assure that the reactor will not be operated with a large number of rods with failed collet housings.

Surveillance Requirements:

The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram, since, if a rod can be moved with drive pressure, it will scram because of higher pressure applied during scram. The frequency of exercising the control rods under the conditions of three or more inoperable rods provides even further assurance of the reliability of the remaining control rods. The checks are done at power levels greater than 30% rated thermal power to clear the RWM and RSCS interlocks.

3.3.C. Control Rod Drive System

1. Control Rod Drive Coupling Integrity

Limiting Conditions for Operation:

Operability of the control rod drive system requires that the drive be coupled to the control rod. In the analysis of control rod drop accidents (FSAR subsection 14.4.3), it has been assumed that one control rod drive coupling has lost its integrity. To assure that not more than one coupling could be in this condition, it is required that either a drive is coupled to the control rod or the drive is fully inserted and disarmed electrically. This requirement serves to maintain operation within the envelope of conditions considered by the plant safety analyses.

Surveillance Requirements:

Observation of a response from the nuclear instrumentation during an attempt to withdraw a control rod provides an indication that the rod is following the drive. The overtravel position feature provides a positive check on the coupling integrity, for only an uncoupled drive can reach the overtravel position.

2. Scram Insertion Times

Limiting Conditions for Operation:

The control rod drive system is designed to bring the reactor subcritical at a rate fast enough to prevent excessive fuel damage. The limiting power transient is that resulting from a loss of condenser vacuum (FSAR Appendix G, Event 12, turbine stop-value closure with closure of the turbine bypass system). Analysis of the transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specification provide the required protection and MCHFR remains greater than 1.0. The limit on the number and pattern of rods permitted to have long scram times is specified to assure that the effect of rods of long scram times are minimized in regard to reactivity insertion rate. Grouping of long scram time rods is prevented by not permitting more than one slow rod in any four rod array. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no operable control rod to have a scram insertion time for 90% insertion greater than 7 seconds.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT TO LICENSE NO. DPR-57

AND

CHANGES TO THE TECHNICAL SPECIFICATIONS

INOPERABLE CONTROL ROD LIMITATIONS

GEORGIA POWER COMPANY

EDWIN I. HATCH UNIT 1

DOCKET NO. 50-321

Introduction

On June 27, 1975, Commonwealth Edison Company (CE) informed NRC that cracks had been discovered on the outside surface of the collet housing of four control rod drives at Dresden Unit 3[1]. The cracks were discovered while performing maintenance of the control rod drives; the reactor was shutdown for refueling and maintenance. In a letter dated July 3, 1975, CE informed us that if the cracks propagated until the collect housing failed, the affected control rod could not be moved[2]. In a meeting with representatives of General Electric (GE) and CE the NRC staff was advised that further inspections revealed cracks in 19 of the 52 Dresden 3 control rod drives inspected, in one spare Dresden 2 control rod drive, in one Vermont Yankee spare control rod drive and in two GE test drives[3]. In a report dated July 30, 1975, after additional rod drives were inspected, CE stated that cracks had been found in 24 of 65 drives inspected[4]. Recently, the Tennessee Valley Authority reported that cracks were found in the collet housing of

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(1) Telegram to J. Keppler, Region III of the NRC, June 27, 1975, Docket No. 50-249.

(2) Letter from B. B. Stephenson, Commonwealth Edison Company to James G. Keppler, U. S. Nuclear Regulatory Commission, July 3, 1975, Docket No. 50-249.

(3) Memo from L. N. Olshan, Division of Technical Review (DTR) to T. M. Novak, DTR, "Meeting on Cracks Found in Dresden 3 Control Rod Drive Collet Retainer Tubes," July 18, 1975.

(4) Letter from B. B. Stephenson, Commonwealth Edison Company to James G. Keppler, U. S. Nuclear Regulatory Commission, July 30, 1975, Docket No. 50-249.

seven of nineteen drives inspected at Browns Ferry 1 and Vermont Yankee found cracks in the collet housing of 4 of 10 control rod drives inspected. Because a number of control rod drives have been affected, because complete failure of the drive collet housing could prevent scram of the affected rod, and because we do not consider existing license requirements adequate in veiw of the collet housing cracks experienced, we have concluded that the Technical Specifications should be changed for those reactors with control rod drive designs susceptible to collet housing cracks. The change should assure that reactors which could be affected would not be operated for extended periods of time with a control rod which cannot be moved.

DESCRIPTION

The control rod drive is a hydraulically operated unit made up primarily of pistons, cylinders and a locking mechanism to hold the movable part of the drive at the desired position. The movable part of the drive includes an index tube with circumferential grooves located six inches apart. The collet assembly which serves as the index tube locking mechanism contains fingers which engage a groove in the index tube when the drive is locked in position. In addition to the collet, the collet assembly includes a return spring, a guide cap, a collet retainer tube (collet housing) and collet piston seals. The collet housing surrounds the collet and spring assembly. The collet housing is a cylinder with an upper section of wall thickness 0.1 inches and a lower section with a wall thickness of about 0.3 inches. The cracks occurred on the outer surface of the upper thin walled section near the change in wall thickness.

1. Consequences of Cracking

The lower edges of the grooves in the index tube are tapered, allowing index tube insertion without mechanically opening the collet fingers, as they can easily spring outward. If the collet housing were to fail completely at the reported crack location, the coil collet spring could force the upper part of the collet housing and spring retainer upward, to a location where the spring and spring retainer would be adjacent to the collet fingers. The clearance between the collet fingers and the spring when in this location will not permit the collet fingers to spring out of the index tube groove. This would lock the index tube in this position so that the control rod could not be inserted or withdrawn.

The failure of some control rods to operate has previously been evaluated and the Technical Specifications presently allow a limited number of rods, as discussed later in Section 4, to be inoperable. If more than these rods are inoperable or if the scram reactivity rate is too small or is shutdown reactivity requirements are not met, the existing Technical Specifications require the reactor to be brought to a cold shutdown condition. Reactor power operation with these rods inoperable would not involve a new hazards consideration nor would it endanger the health and safety of the public.

2. Probable Cause of Cracking.

The cause of the cracking appears to be a combination of thermal cycling and intergranular stress corrosion cracking. The thermal cycling results from insertion and scram movements. During these movements hot reactor water is forced down along the outside of the collet housing, while cool water is flowing up the inside and out of flow holes in the housing. These thermal cycles are severe enough to yield the material, leaving a high residual tensile stress on the outer surface.

The collet housing material is type 304 austenitic stainless steel. The lower portion of the collet housing has a thicker wall and its inner surface is nitrided for wear resistance. In 1960-61, similar drives using high hardness 17-4 PH material for index tubes and other parts were found to have developed cracks. The problem caused GE to switch to nitrided stainless steel. The nitriding process involves a heat treatment in the 1050 F to 1100 F range, which sensitizes the entire collet housing, making it susceptible to oxygen stress corrosion cracking.

The cooling water used in the drives is aerated water. This water contains sufficient oxygen for stress corrosion to occur in the sensitized material if it is subjected to the proper combination of high stresses and elevated temperatures.

We believe that the cracking is caused by a combination of thermal fatigue and stress corrosion. GE has determined that both full stroke insertion and scram will cause high thermal stress. The cracks are completely intergranular and extensively branched, indicating that corrosion is a major factor. The type of thermal cycling, plus the buildup of corrosion products in the cracks between cycles probably results in a ratcheting action. This is also indicated by the "bulged" appearance of the cracks on the OD.

3. Probability of Early Failure

We belive that the cracking is progressive and is cycle dependent. Although the details of the cracking process are still not clear, we have not identified any mechanism that would cause rapid cracking with progression to complete circumferential failure.

The axial loads on the housings are very low at all times so that through wall cracks would have to progress at least 90% around the circumference before there would be concern about a circumferential failure. Although one housing at Dresden 3 had three cracks which nearly joined around the circumference, no cracks at Dresden 3 were through wall and none of the housing examined approached the degree of cracking necessary for failure. The collet housing has three flow holes in the thin section equally spaced around the circumference. The observed cracks have been confined primarily to the areas below and between the holes and near the area where the wall thickness of the collet housing changes. Since all the cracks except those located at the change in wall thickness are fairly shallow and since those at the change in wall thickness are largely confined to the circumferential area between holes, the net strength of the cracked housings is still far greater than necessary to perform their function.

A test drive at GE that had experienced over 4000 scram cycles had a more extensive developed crack pattern. Although the satisfactory experience with this cracked test housing is encouraging, its performance may not be correlated directly to that of drives in service, as this test drive was subjected to lower temperatures, and possibly less severe thermal cycles than could be encountered in actual service. The cracks were first noticed on the test drive after about 2000 cycles - many more cycles than the cracked housings at Dresden 3 had experienced.

The chance that a large number of collet housing would fail completely at about the same time is very remote. This is primarily true because the distributions of failures by cracking mechanisms such as stress corrosion and fatigue are not linear functions. That is, failure is a function of log time or log cycles. Distribution of failures of similar specimens generally follow a log normal pattern, with one or two orders of magnitude in time or cycles between failures of the first and failures of the last specimen. As no collet housing has yet failed, we are confident that there would be very few, if any, failures during the next time period corresponding to the total service life to date.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY AND

OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

NOTICE OF PROPOSED ISSUANCE OF AMENDMENT
TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendment to Facility Operating License No. DPR-57 issued to Georgia Power Company & Oglethorpe Electric Membership Corporation for operation of the Edwin I. Hatch Nuclear Plant Unit 1 located in Appling County, Georgia.

This amendment would revise the Technical Specifications to add requirements that would limit the period of time operation can be continued with immovable control rods that could have control rod mechanism collect housing failures.

Prior to issuance of the proposed license amendment, the Commission will have made the findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations.

By 19 Jan. 1976, the licensee may file a request for a hearing and any person whose interest may be affected by this proceeding may file a request for a hearing in the form of a petition for leave to intervene with respect to the issuance of this amendment to the subject facility operating license. Petitions for leave to intervene must be filed under oath or affirmation in accordance with the provisions of Section 2.714 of

10 CFR Part 2 of the Commission's regulations. A petition for leave to intervene must set forth the interest of the petitioner in the proceeding, how that interest may be affected by the results of the proceeding, and the petitioner's contentions with respect to the proposed licensing action. Such petitions must be filed in accordance with the provisions of this FEDERAL REGISTER notice and Section 2.714, and must be filed with the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Section, by the above date. A copy of the petition and/or request for a hearing should be sent to the Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555 and to G. F. Trowbridge, Esquire, Shaw, Pittman, Potts and Trowbridge, Barr Building, 910 17th Street, N. W., Washington, D. C. 20006, attorney for the licensee.

A petition for leave to intervene must be accompanied by a supporting affidavit which identifies the specific aspect or aspects of the proceeding as to which intervention is desired and specifies with particularity the facts on which the petitioner relies as to both his interest and his contentions with regard to each aspect on which intervention is requested. Petitions stating contentions relating only to matters outside the Commission's jurisdiction will be denied.

All petitions will be acted upon by the Commission or licensing board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel. Timely petitions will be considered to determine whether a hearing should be noticed or another appropriate order issued.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

regarding the disposition of the petitions.

In the event that a hearing is held and a person is permitted to intervene, he becomes a party to the proceeding and has a right to participate fully in the conduct of the hearing. For example, he may present evidence and examine and cross-examine witnesses.

For further details with respect to these actions, see the Commission's letter to Georgia Power Company & Oglethorpe Electric Membership Corporation dated September 23, 1975 and the attached proposed Technical Specifications and the Safety Evaluation by the Commission's staff dated September 23, 1975 and Georgia Power Company's letter dated October 13, 1975, which 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. This license amendment and the Safety Evaluation may be inspected at the above locations and a copy 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 day of

FOR THE NUCLEAR REGULATORY COMMISSION

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

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DATE ➤	12/ 3 /75	12/ 4 /75	12/ 4 /75			