Central Files

March 16, 1976

Docket No. 50-324

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
Raleigh, North Carolina 27603

Gentlemen:

The Commission has issued the enclosed Amendment No.11 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant Unit 2. This amendment consists of changes to the Technical Specifications, and is based on our letters to you dated September 23, and December 15, 1975, and your response dated October 13, 1975.

In your letter of October 13, 1975, and in discussions with your staff, you raised several objections to our proposed course of action as stated in our letter of September 23, 1975. We intend to further consider your comments in light of information to be submitted by the General Electric Company. At the present time, however, we consider the incidence of cracked control rod drive collet housings at operating BWRs to be of sufficient importance as to warrant this course of action. Conversations held with your staff indicated that you would accept the issuance of the enclosed license amendment.

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

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

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Please note that we have discontinued the use of separate identifying numbers for changes to technical specifications. Sequential amendment numbers will be continued as in the past.

Sincerely,

C. M. Trammell

for/ Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 11
2. Safety Evaluation
3. Federal Register Notice

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See next page

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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. DPR-62

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

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated October 13, 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;

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

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-62 is hereby amended to read as follows:

"2.C.(2) 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."

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

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 16, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 11

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Revise Appendix A as follows:

Remove pages:	3.3-1/3.3-2 3.3-9/3.3-10
Insert new pages:	3.3-1/3.3-1a 3.3-1b/3.3-2 3.3-9/3.3-10 3.3-10a/3.3-10b

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.3 Reactivity Control

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

   The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operating cycle with the strongest operable control rod in its full-out position and all other operable rods fully inserted.

4.3 Reactivity Control

Applicability:

Applies to the surveillance requirements of the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

   Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.28 percent Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

Amendment No. 11
March 16, 1976

2. <u>Reactivity margin - inoperable control rods</u>

a. Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. 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.A.2.b. If investigation demonstrates that the cause of the control rod drive failure is a cracked collet housing, or if this possibility cannot be ruled out, the reactor shall not be started until the affected control rod drive has been replaced or repaired.

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically and the control rods shall be in such positions that Specification 3.3.A.1 is met.

2. <u>Reactivity margin - inoperable control rods</u>

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 20% 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 20% power.

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LIMITING CONDITIONS FOR OPERATION

3.3.A Reactivity Limitations (Cont'd)

c. Control rod drives which are fully inserted and electrically disarmed shall not be considered inoperable.

d. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

e. During reactor power operation, the number of inoperable control rods shall not exceed eight. In addition, 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). Specification 3.3.A.1 must be met at all times.

B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

SURVEILLANCE REQUIREMENTS

4.3.A Reactivity Limitations (Cont'd)

b. When it is initially determined that a control rod is incapable of normal insertions, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted, a shutdown margin test shall be made to demonstrate 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.

B. Control Rods

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

a. When the rod is withdrawn the first time subsequent to each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is critical shall be performed to verify instrumentation response.

BASES:

3.3 and 4.3 Reactivity Control

A. Reactivity Limitation

1. The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.28 percent Δk at the time of the test, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in percent Δk is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., the initial loading. R must be a positive quantity or zero. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter.

   The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. A new value of R must be determined for each fuel cycle.

   The 0.28 percent Δk in the expression R + 0.28 percent Δk is provided as a finite, demonstrable, subcriticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.28 percent Δk in reactivity. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but at least an R $\pm$ 0.28 percent Δk margin beyond this.

2. Reactivity margin - inoperable control rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and then 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.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest operable control rod does not insert. An allowable pattern for control rods valved out of service, which shall meet this Specification, will be determined and made available to the operator. The number of rods permitted to be inoperable could be many more than the eight allowed by the Specification, particularly late in the operation cycle; however, the occurrence of more than eight could be indicative of a generic control rod drive problem and the reactor will be shut down. Also, if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from

* 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 encrusted accumulations in the drive. Electrical disarming does not eliminate position indication.

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.

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Amendment no. 11
March 16, 1976

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 11 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER AND LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-324

INTRODUCTION

On June 27, 1975, Commonwealth Edison Company (CE) informed NRC that cracks had been discovered on the outside surface of the collet housings 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 collet 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 Compa James G. Keppler, U. S. Nuclear Regulatory Commission, 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 view 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 up to eight control rods to operate has previously been evaluated and the Technical Specifications presently allow up to eight rods to be inoperable. If more than eight rods are inoperable or if the scram reactivity rate is too small or if 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 up to eight 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 believe 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 housings 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 to 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.

4. Changes to Technical Specifications

Existing limiting conditions of operation allow operation to continue with up to one inoperable control rod in any 5 x 5 array. Existing surveillance requirements specify that daily surveillance of the condition of all fully or partially withdrawn rods would not have to begin until three rods are found inoperable. The surveillance requirements also specify that if it is determined that a control rod cannot be inserted, the reactor shall be brought to a Cold Shutdown Condition within 24 hours to perform a shutdown margin test. If the shutdown margin requirements are determined to be met the reactor may be returned to operation with the rod which is incapable of being inserted. We do not consider that these existing requirements sufficiently limit the possibility of operating for an extended period of time with a number of rod drive mechanisms which cannot be moved. We have therefore concluded that the Technical Specifications should be changed as discussed below.

> One stuck control rod does not create a significant safety concern. However, if a rod cannot be moved and the cause of the failure cannot be determined, the rod could have a failed collet housing. A potentially failed collet housing would be indicative of a problem which could eventually affect the scram capability of more than one control rod. Since the cracks appear to be of a type which propagate slowly, it is highly unlikely that a second control rod would experience a failed collet housing within a short period of time after the first failure. Therefore, Section 3.3.A.2 (Reactivity Margin - Inoperable Control Rods) should be expanded to preclude reactor startup and/or continued power operation with a partially or fully withdrawn control rod which cannot be moved with drive or scram pressure, 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. If investigation demonstrates that the cause of the control rod drive failure is a cracked collet housing, or if this possibility cannot be ruled out, the reactor should not be started until the affected control rod drive has been repaired or replaced.

Until permanent corrective measures are taken to resolve the potential for stuck control rods due to failed collet housings, we believe that these additional specifications provide reasonable assurance that an unacceptable number of control rod collet housing will not fail during operation. Upon completion of the investigations being performed by GE, additional corrective actions may permit revision of these requirements.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above that: (1) 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: March 16, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-324

CAROLINA POWER AND LIGHT COMPANY

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. 11to Facility Operating License No. DPR-62 issued to the Carolina Power and Light Company (the licensee), which revised Technical Specifications for operation of the Brunswick Steam Electric Plant, Unit 2 (the facility), located in Brunswick County, North Carolina. The amendment is effective as of the date of issuance.

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

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 the Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on December 23, 1975 (40 FR 59379). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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DATE ➤						

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

For further details with respect to this action, see (1) the Commission's letters to Carolina Power and Light Company dated September 23, 1975 and December 15, 1975, (2) Amendment No. 11 to License No. DPR-62, 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 Southport Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

A single copy of items (1), (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 16th day of March 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Trammell, Acting Chief
Operating Reactors Branch #1
Division of Operating Reactors

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