Docket No. 50-278

November 29, 1993

Mr. George A. Hunger, Jr.
Director-Licensing, MC 52A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

SUBJECT: EMERGENCY TECHNICAL SPECIFICATION CHANGE TO ALLOW OPERATION OF CONTROL ROD 54-35 UNCOUPLED FROM ITS DRIVE FOR REMAINDER OF CYCLE 10, PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3 (TAC NO. M88236)

The Commission has issued the enclosed Amendment No. 187 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Unit No. 3. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 22, 1993. This was prepared and issued on an emergency basis to avoid a forced shutdown. A shutdown would be required to effect repairs on the control rod drive in order to avoid premature burnup of four control rods and derating of cycle capacity and energy output.

This amendment revises TS 3.3.B.1 to allow control rod 54-35 to be uncoupled for the remainder of cycle 10 (to be completed before 10/30/95). The amendment specifies conditions under which rod 54-35 may be operated to verify rod position by use of neutron instrumentation.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance of Amendment to Facility Operating License and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,
Original signed by:
Jose A. Calvo, Assistant Director
for Region I Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 187 to License No. DPR-56
2. Safety Evaluation

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 29. 1993

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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. George A. Hunger, Jr.
Philadelphia Electric Company

Peach Bottom Atomic Power Station,
Units 2 and 3

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PHILADELPHIA ELECTRIC COMPANY

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 187
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, et. al. (the licensee) dated November 22, 1993, 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-56 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 187 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

José A. Calvo

Jose A. Calvo, Assistant Director
for Region I Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: November 29. 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 187

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following page of the Appendix A Technical Specifications with the enclosed page. The revised area is indicated by a marginal line.

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LIMITING CONDITIONS FOR OPERATION

3.3.A Reactivity Limitations
(Cont'd)

f. Inoperable control rods shall be positioned such that specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5 x 5 array may be operable (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 cold shutdown condition within 24 hours.

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 except as in 3.3.B.1.a. 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.

   a. For control rod 54-35, for the remainder of cycle 10 (to be completed before 10/30/95).

      If coupling cannot be accomplished, the uncoupled control rod may be withdrawn when ≥10% of rated thermal power only if all the following conditions are satisfied:

      1) no other uncoupled control rod is withdrawn;

      2) the uncoupled control rod may not be withdrawn past notch position 46.

SURVEILLANCE REQUIREMENTS

4.3.A Reactivity Limitations
(Cont'd)

B. Control Rods

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

   a. When a rod is withdrawn the first time after each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation and rod position indication for 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 the Rod Worth Minimizer low power setpoint shall be performed to verify instrumentation response.

   b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance observe that the drive does not go to the overtravel position.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 187 TO FACILITY OPERATING LICENSE NO. DPR-56

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

DOCKET NO. 50-278

1.0 INTRODUCTION

By letter dated November 22, 1993, the Philadelphia Electric Company (the licensee) submitted a request for changes to the Peach Bottom Atomic Power Station, Unit No. 3, Technical Specifications (TS). The requested changes would revise TS 3.3.B.1.a to refer to control rod 54-35, for the remainder of cycle 10 (to be completed before 10/30/95). The requested change would permit operation of the facility with the control rod uncoupled for the remainder of the operating cycle. Neutron monitoring by means of either the Local Power Range Monitor (LPRM) or Transversing Incore Probe (TIP) Systems would be used to verify the control rod movement. This amendment request is similar to the licensee's June 14, 1991 request for control rod 38-23. The staff granted that request by Amendment No. 166 dated July 10, 1991.

2.0 EVALUATION

The primary concern for control rod coupling integrity is its impact on the potential increase in the probability of a control rod drop accident (CRDA) as analyzed in the Final Safety Analysis Report (FSAR). Additionally, control rod coupling integrity ensures that indicated control rod position is indicative of actual control rod position. The uncoupled rod condition also raises an operational concern for equipment damage due to scram loading. The rod could separate from the control rod drive (CRD) during the deceleration phase of the scram stroke which could result in increased loads on the affected parts.

In its November 22, 1993 submittal, the licensee addressed each of the concerns identified above:

For the CRDA concern, the licensee stated that above 10% of the rated thermal power, the consequences of a CRDA are negligible; therefore, the control rod coupling integrity will not increase the consequences of a CRDA (the revised TS requires that control rod 54-35 be fully inserted below 10% power).

Regarding the ability to verify the rod's actual position, the licensee stated that the neutron monitoring systems of the LPRM and TIP systems

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will verify the position of the control rod by providing an indication that the control rod is following the CRD indication and the control rod is not stuck in the reactor.

Finally, the licensee addressed the possibility of control rod damage due to scram loading. The licensee stated that the imposed limitation on the control rod not to be withdrawn past position "46" will minimize the mechanical loads on the control rod to CRD spud/socket coupling mechanism.

Regarding the licensee's responses delineated above, the staff agrees that CRDAs are not a concern at or above 10% power. The staff also agrees that neutron instruments can be utilized to verify actual rod motion. Regarding scram loading, in a July 10, 1991 letter, the staff previously reviewed and approved a similar licensee proposal to minimize the mechanical loads during a scram. The staff has reviewed its original finding and determined that it is still appropriate.

Based on the staff's review of the licensee's submittal, the licensee has satisfied all of the staff's concerns regarding uncoupled control rods. The staff believes that the reactor operators will be able to adequately perform the compensatory measures delineated in the revised TS for a single uncoupled rod. Therefore, the staff has determined that the proposed changes satisfy staff positions and requirements in these areas. Operation with control rod 54-35 withdrawn in accordance with the proposed procedures and TS is acceptable.

3.0 EMERGENCY CIRCUMSTANCES

In its submittal, the licensee stated that the condition of the uncoupled CRD was identified just prior to returning to power operation from the ninth refueling outage. At that time, the licensee determined that there were no adverse safety consequences associated with operating through the next fuel cycle with an uncoupled CRD. The rod was then declared inoperable, fully inserted and electrically disarmed in accordance with the TS. The licensee determined that commencing with the startup allowed them to identify any other conditions that would require shut down and repairs. Since there were no other significant difficulties noted, the licensee proceeded with the startup. However, it was unknown to the licensee at that time that a forced shutdown would be required to affect repairs on this CRD in order to avoid premature burnup of the four control rods and derating of cycle capacity and energy output.

The licensee stated that because of the age and the exposure history of the four control rods, full power operation after December 6, 1993 will require that during the next refueling outage these blades be replaced. Operation up to December 6, 1993 will result in the blades reaching the cracking threshold limit of 20% B-10 specified in GE Service Information Letter 157. The blades may be able to be reused by minimizing their exposure while in the core; however, this would require a significant derating of the reactor.

The staff has reviewed the circumstances associated with the licensee's request for an emergency TS change and has concluded that failure to act in a timely manner would result in subsequent derating of the plant. The staff has concluded that this condition could not have reasonably been foreseen due to the unique circumstances that result from fully inserting control rods in different core locations. Therefore, this amendment is being processed under 10 CFR 50.91(a)(5).

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The licensee proposed that the TS change did not involve a significant hazards consideration, stating as follows:

> "The following evaluation is provided in accordance with the requirements of 10 CFR 50.92:
>
> 1. Does the proposed amendment involve a significant increase in the probability of consequences of an accident previously evaluated?
>
>   No, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. This amendment incorporates compensatory actions in the Technical Specifications to assure that even with an uncoupled rod the rod position is known, that no other uncoupled rods are withdrawn, and that scram performance remains intact. [As stated in the "Safety Discussion" section of the licensee's submittal, a CRDA was the accident evaluated. The licensee stated that above 10% of the rated thermal power, the consequences of a CRDA are negligible; therefore, the control rod coupling integrity will not increase the consequences of a CRDA because the revised TS requires that control rod 54-35 be fully inserted below 10% power.]
>
> 2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?
>
>   No, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated. The

compensatory measures included in the Technical Specification changes assure that no new or different kind of accident is possible.

3. <u>Does the proposed amendment involve a significant reduction in the margin of safety?</u>

   No, the proposed amendment does not involve a significant reduction in the margin of safety as the limiting event is the CRDA and all fuel limits stipulated in that analysis will be met when the compensatory measures included in Technical Specification changes are implemented.

Based on the above discussion, and the compensatory measures evaluated in Section 2.0 above, the staff concludes that this amendment meets the criteria and therefore, does not involve a significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. In a November 23, 1993 telephone call, the state official said that he would have no comments regarding the safety aspects of this proposed amendment. However, he questioned whether the licensee should have made an attempt to repair the CRD as soon as it became apparent that there was an uncoupled control rod. He said he would understand not immediately attempting to repair the CRD if it would result in a significant delay in return to operation. To address the state official's concern, the staff telephoned the licensee on November 24, 1993. The licensee stated that stopping to repair the CRD would have delayed the startup by approximately 5 days. Also, commencing with the start up allowed the licensee to identify any other conditions that would require shut down and repairs. The staff believes that the licensee's approach was reasonable, considering the information the licensee had at the time the decision was made to proceed with the startup.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards finding with respect to this amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) significantly increase the probability or consequences of an accident previously evaluated, (b) increase the possibility of a new or different kind of accident from any previously evaluated or (c) significantly reduce a safety margin and, therfore, 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; (3) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: S. Dembek

Date: November 29. 1993