

January 16, 1998

Mr. George A. Hunger, Jr.  
Director-Licensing, MC 62A-1  
PECO Energy Company  
Nuclear Group Headquarters  
Correspondence Control Desk  
P.O. Box No. 195  
Wayne, PA 19087-0195

SUBJECT: REVISION TO TECHNICAL SPECIFICATIONS REGARDING CONTROL ROD 50-27,  
LIMERICK GENERATING STATION, UNIT 1 (TAC NOS. M99854 AND M99855)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendment No. 124 to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 24, 1997.

This amendment changes Sections 3.1.3.6 and 4.1.3.6 of the LGS Unit 1 TS to allow operation of control rod 50-27, uncoupled from its drive, for the remainder of Cycle 7. The amendment specifies conditions under which control rod 50-27 may be operated and modifies existing surveillance requirements to verify control rod position by use of neutron instrumentation.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

This completes our efforts on this issue and therefore, we are closing out TAC Nos. M99854 and M99855.

Sincerely,  
Original signed by:  
Bartholomew C. Buckley, Senior Project Manager  
Project Directorate I-2  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Docket No. 50-352

- Enclosures: 1. Amendment No. 124 to License No. NPF-39  
2. Safety Evaluation

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

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Sincerely,

A handwritten signature in cursive script that reads "Bartholomew C. Buckley".

Bartholomew C. Buckley, Senior Project Manager  
Project Directorate I-2  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Docket No. 50-352

Enclosures: 1. Amendment No. 124 to  
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2. Safety Evaluation

cc w/encls: See next page

Mr. George A. Hunger, Jr.  
PECO Energy Company

Limerick Generating Station,  
Units 1 & 2

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

PHILADELPHIA ELECTRIC COMPANY  
DOCKET NO. 50-352  
LIMERICK GENERATING STATION, UNIT 1  
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124  
License No. NPF-39

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Philadelphia Electric Company (the licensee) dated October 24, 1997, 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.

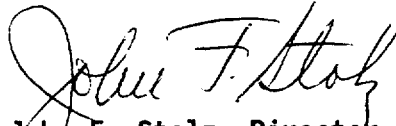
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-39 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 124, are hereby incorporated in the license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance, and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



John F. Stolz, Director  
Project Directorate I-2  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical  
Specifications

Date of Issuance: January 16, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 124

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove

3/4 1-11

3/4 1-12

Insert

3/4 1-11

3/4 1-12

REACTIVITY CONTROL SYSTEMS  
CONTROL ROD DRIVE COUPLING  
LIMITING CONDITION FOR OPERATION

---

3.1.3.6 All control rods shall be coupled to their drive mechanisms.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5\*.

ACTION:

- a. In OPERATIONAL CONDITIONS 1 and 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:
  1. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:
    - a) Observing any indicated response of the nuclear instrumentation, and
    - b) Demonstrating that the control rod will not go to the over-travel position.Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.
  2. If recoupling is not accomplished on the first attempt or, if not permitted by the RWM, then except as in 3.1.3.6.d or until permitted by the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves\*\* either:
    - a) Electrically, or
    - b) Hydraulically by closing the drive water and exhaust water isolation valves.Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.
- b. In OPERATIONAL CONDITION 5\* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours either:
  1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or
  2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves\*\* either:
    - a) Electrically, or
    - b) Hydraulically by closing the drive water and exhaust water isolation valves.
- c. The provisions of Specification 3.0.4 are not applicable.
- d. For control rod 50-27, for the remainder of Unit 1 Cycle 7, if coupling can not be established the uncoupled rod may be withdrawn when rated thermal power exceeds 10% only if all the following conditions are satisfied:
  - 1) The uncoupled control rod may not be withdrawn past notch position 46, and
  - 2) No other uncoupled control rod is withdrawn.

---

\* At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

\*\* May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

## REACTIVITY CONTROL SYSTEMS

### SURVEILLANCE REQUIREMENTS

---

4.1.3.6 Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

- a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,
- b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation,
- c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity, and
- d. When repositioning the uncoupled control rod per Specification 3.1.3.6.d the uncoupled rod's position shall be verified to have followed the control rod drive by neutron instrumentation (LPRM or TIP). If the control blade can not be verified to have followed the drive out to its final position, then the rod shall be completely inserted and the control rod directional valves disarmed as stated in 3.1.3.6.a.2.





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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
RELATED TO AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. NPF-57

PHILADELPHIA ELECTRIC COMPANY  
LIMERICK GENERATING STATION, UNIT 1

DOCKET NO. 50-352

1.0 INTRODUCTION

By letter dated October 24, 1997, Philadelphia Electric Company (the licensee) submitted a request for changes to the Limerick Generating Station (LGS), Unit 1, Technical Specifications (TSs). The requested changes would permit operation of the facility with control rod 50-27 uncoupled for the remainder of Cycle 7 for LGS Unit 1. Specifically, the change would revise Section 3/4.1.3.6 of the LGS Unit 1 TS to exempt control rod 50-27 from the coupling test for the remainder of Cycle 7, provided certain conditions are met. Also, repositioning of the control rod to its proposed step 46 position would be allowed when the reactor power is above 10% of rated power. The surveillance requirements of TS 4.1.3.6.d would include additional monitoring during the repositioning evolution. In this regard, neutron monitoring by means of either the Local Power Range Monitor (LPRM) or the Traversing Incore Probe (TIP) systems would be used to verify the control rod movement.

On August 31, 1997, during the weekly exercise test of control blade 50-27, when the blade was inserted one notch from the full-out position 48 to position 46 and then returned to full out, the full-out position indication of 48 was lost. A continuous withdrawal signal was selected to restore position indication which resulted in a "rod overtravel" alarm, indicating decoupling of the drive from the control blade. In accordance with existing procedures, power was reduced to below 60%, followed by full insertion of blade 50-27 to position 00 and its being disarmed. Upon subsequent troubleshooting, the blade was fully withdrawn (again, without receiving position indication for the full-out position of 48). Blade movement was verified during the withdrawal by monitoring LPRMs. The blade was judged to be coupled since an overtravel alarm was not received following selection of a continuous withdrawal signal after the blade was judged to be fully withdrawn; however, since the full-out position indication of 48 was not received, the blade could not procedurally be determined to be coupled and was therefore re-inserted fully to position 00 and disabled by hydraulic block in accordance with TS requirements.

Limiting Condition for Operation 3.1.3.6.a.2. requires that if recoupling is not accomplished in the first attempt, the rod is to be declared inoperable, fully inserted, and disabled electrically or hydraulically. Surveillance Requirement 4.1.3.6.b. requires that the coupling test be performed any time the control rod is withdrawn to the full out position. The only way to verify rod coupling,

as described in existing plant procedures (Control Rod Exercise and Control Rod Coupling Check) is to fully withdraw the blade to position 48, and then apply a continuous withdrawal signal to verify that no overtravel alarm occurs. By the nature of the problem with blade 50-27, an indication of 48 cannot be received; hence, it cannot be positively ascertained that the blade is fully out in order to apply the withdrawal signal to check for overtravel. Since the blade can not be assured of being coupled, it must be assumed to be not coupled. The proposed TS change would suspend the requirement to fully insert and disarm the subject blade for the remainder of Cycle 7 at LGS Unit 1.

## 2.0 PROPOSED TS CHANGES

### 2.1 Limiting Condition for Operation 3.1.3.6.a.2. has the underlined phrase added:

If recoupling is not accomplished on the first attempt or, if not permitted by the RWM, then except as in 3.1.3.6.d or until permitted by the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves\*\* either:

(The balance of this paragraph remains unchanged.)

### 2.2 Limiting Condition for Operation action 3.1.3.6.d. is added:

- d. For control rod 50-27, for the remainder of Unit 1 Cycle 7, if coupling can not be established the uncoupled rod may be withdrawn when rated thermal power exceeds 10% only if all the following conditions are satisfied:
  - 1) The uncoupled control rod may not be withdrawn past notch position 46, and
  - 2) No other uncoupled control rod is withdrawn.

### 2.3 Surveillance Requirement 4.1.3.6.d. is added:

- d. When repositioning the uncoupled control rod per Specification 3.1.3.6.d the uncoupled rod's position shall be verified to have followed the control rod drive by neutron instrumentation (LPRM or TIP). If the control blade can not be verified to have followed the drive out to its final position, then the rod shall be completely inserted and the control rod directional valves disarmed as stated in 3.1.3.6.a.2.

## 3.0 EVALUATION

The licensee's attempt to verify that the coupling integrity of rod 50-27 was unsuccessful, and, therefore, it must be assumed that the control rod and drive are uncoupled. 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. Additionally, control rod coupling integrity ensures that the indicated control rod position is indicative of the 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.

The licensee has proposed procedural changes to assure that withdrawal operation with the uncoupled rod 50-27 will not pose a significant CRDA concern for the remainder of the fuel Cycle 7. The rod will remain inserted and not be withdrawn at reactor power levels less than 10 percent. During the withdrawal sequence above 10 percent, neutron flux information in the vicinity of the rod will be monitored to verify that the control rod blade tracks with the drive movement. This will ensure that the rod is not sticking and separated from the CRD.

An analysis performed by General Electric Company (GE) for the licensee shows that, for the fuel cycle under consideration, the consequences of a CRDA at power levels above 10 percent of rated thermal power are negligible and that no constraints on control rod sequences are required. Below 10 percent of rated power, the uncoupled control rod will be fully inserted. Above 10 percent power, the compensatory actions ensure that the Rod Block Monitor mitigates the consequences of a Rod Withdrawal error.

The licensee's proposed changes provide an additional measure to minimize the possibility of a CRDA by requiring the use of neutron instrumentation (LRPM or TIP) to verify rod position during repositioning of the uncoupled rod. This is addressed in the proposed change to Surveillance Requirement 4.1.3.6.d.

The GE analysis also addressed the possibility of equipment damage from scram loadings. Mechanism damage could occur during the deceleration phase of the scram stroke. The uncoupled rod would continue to move upward and the velocity limiter would strike the bottom of the fuel support casting. Analysis shows that in this scenario, damage might occur to the velocity limiter, or upon rebound, to the spud and the lock plug. However, there is insufficient energy to dislodge the fuel support and fuel.

GE has provided recommended operating strategies to minimize possible scram load problems. The recommended operation with rod 50-27 withdrawal limited to notch position 46 minimizes the scram loadings on the spud and socket. The weekly tests of rod movement required by the TSs will continue, thus assuring rod movement capability.

GE has evaluated the effect of the control rod coupling integrity on scram performance. The scram and insertion performance were not considered to be degraded nor would other reactivity control functions be adversely affected. Since the rod will be operated at a slightly inserted position for full withdrawal, it should have slightly better scram reactivity insertion characteristics. It is therefore reasonable to conclude that operation with rod 50-27 fully withdrawn will not lead to any condition adverse to reactor safety.

The TS changes accompanying this mode of operation consist of changes to Sections 3.1.3.6 and 4.1.3.6. With the change to Section 3.1.3.6, rod 50-27 may be withdrawn when rated thermal power is greater or equal to 10 percent under certain conditions. These conditions are that no other uncoupled rods are withdrawn and rod 50-27 may not be withdrawn past notch position 46. The change to Section 4.1.3.6 requires the use of neutron instrumentation (LPRM or TIP) to verify that rod 50-27 followed the CDR during repositioning. If the blade cannot be verified to have followed the drive to its final position, then the rod shall be completely inserted and the control rod directional valves disarmed, as stated in Section 3.1.3.6.a.2. These TS changes adequately implement the required changes in rod operation and are acceptable.

#### 4.0 SUMMARY CONCLUSIONS

Based on the NRC staff review of the licensee's submittal in support of the proposed TS changes in the operation of control rod 50-27 for the remainder of Cycle 7 for LGS Unit 1, we find the proposed amendment acceptable.

#### 5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

#### 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 and changes the surveillance requirements. 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 previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 61844). 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) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) 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 Contributors: H. Richings  
B. Buckley

Date: January 16, 1998