March 19, 1985

Docket No. 50-277

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Mr. Edward G. Bauer, Jr.
Vice President and General Counsel
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Dear Mr. Bauer:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENT TO PERMIT RELOADING AND OPERATING PEACH BOTTOM, UNIT 2, FOR CYCLE 7

The Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. This amendment revises the Technical Specifications (TSs) in response to your application dated September 7, 1984.

The changes to the TSs permit reactor operation of Peach Bottom Unit No. 2 with Reload Number 6 (Cycle 7).

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

Sincerely,

Original signed by

Gerald E. Gears, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 108
2. Safety Evaluation

cc w/enclosure:
See next page

| ORB#4:DL | ORB#4:DL | ORB#4:DL | OELD | AD:OR:DL |
|---|---|---|---|---|
| RIngram | GGears;cr | JStolz | LRFinkelstein | GLainas |
| 3/14/85 | 3/14/85 | 3/15/85 | 3/12/85 | 3/14/85 |

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P PDR

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

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, et al. (the licensee) dated September 7, 1984, 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-44 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.108, 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

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 19, 1985

ATTACHMENT TO LICENSE AMENDMENT NO.108

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

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

| Remove | Insert |
|---|---|
| iv | iv |
| 119 | 119 |
| 133a | 133a |
| 133d | 133d |
| 133e | 133e |
| 142 | --- |
| 142a | 142a |
| 142b | --- |
| 142e | --- |
| 142f | --- |
| 142i | 142i |
| --- | 142k |
| 241 | 241 |

## LIST OF FIGURES

## 3.4 BASES

### STANDBY LIQUID CONTROL SYSTEM

A. The conditions under which the Standby Liquid Control System must provide shutdown capability are identified via the Plant Nuclear Safety Operational Analysis (Appendix G). If no more than one operable control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the Standby Liquid Control system is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the liquid control system is required.

The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration of 660 ppm of boron in the reactor core in less than 125 minutes. The 660 ppm concentration in the reactor core will bring the reactor from full power to a subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

The minimum limitation on the relief valve setting is intended to prevent the recycling of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve setting provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the remaining system will perform its intended function and that the long term average availability of the system is not reduced is obtained for a one out of two system by an allowable equipment out of service time of one third of the normal surveillance frequency. This method determines an equipment out of service time of ten days. Additional conservatism is introduced by reducing the allowable out of service time to seven days, and by increased testing of the operable redundant component.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.5.I Average Planar LHGR | 4.5.I Average Planar LHGR |
| During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in the applicable figures during two recirculation loop operations. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the following reduction factor: 0.79 for P8X8R and BP8X8R fuel. If at any time during operation it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore APLHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. | The APLHGR for each type of fuel as a function of average planar exposure shall be checked daily during reactor operation at $\geq$25% rated thermal power. |
| 3.5.J Local LHGR | 4.5.J Local LHGR |
| During power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed design LHGR.<br><br>LHGR $\leq$ LHGRd<br><br>LHGRd = Design LHGR<br>13.4 kW/ft for all 8X8 fuel | The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power. |

Table 3.5.K.2

OPERATING LIMIT MCPR VALUES
FOR VARIOUS CORE EXPOSURES*

| Fuel Type | MCPR Operating Limit** For Incremental Cycle Core Average Exposure | |
|---|---|---|
| | BOC to 2000 MWD/t Before EOC | 2000 MWD/t before EOC To EOC |
| P8X8R *** | 1.23 | 1.29 |

* If requirement 4.5.K.2.a is met.

** These values shall be increased by 0.01 for single loop operation.

*** Applicable to all P8X8R fuel bundles including BP8X8R and the P8DRB285 (Reload 5) types.

Table 3.5.K.3

OPERATING LIMIT MCPR VALUES
FOR VARIOUS CORE EXPOSURES*

| Fuel Type | MCPR Operating Limit** For Incremental Cycle Core Average Exposure | |
|---|---|---|
| | BOC to 2000 MWD/t Before EOC | 2000 MWD/t before EOC To EOC |
| P8X8R*** | 1.34 | 1.41 |

* If surveillance requirement 4.5.K.2 is not performed.

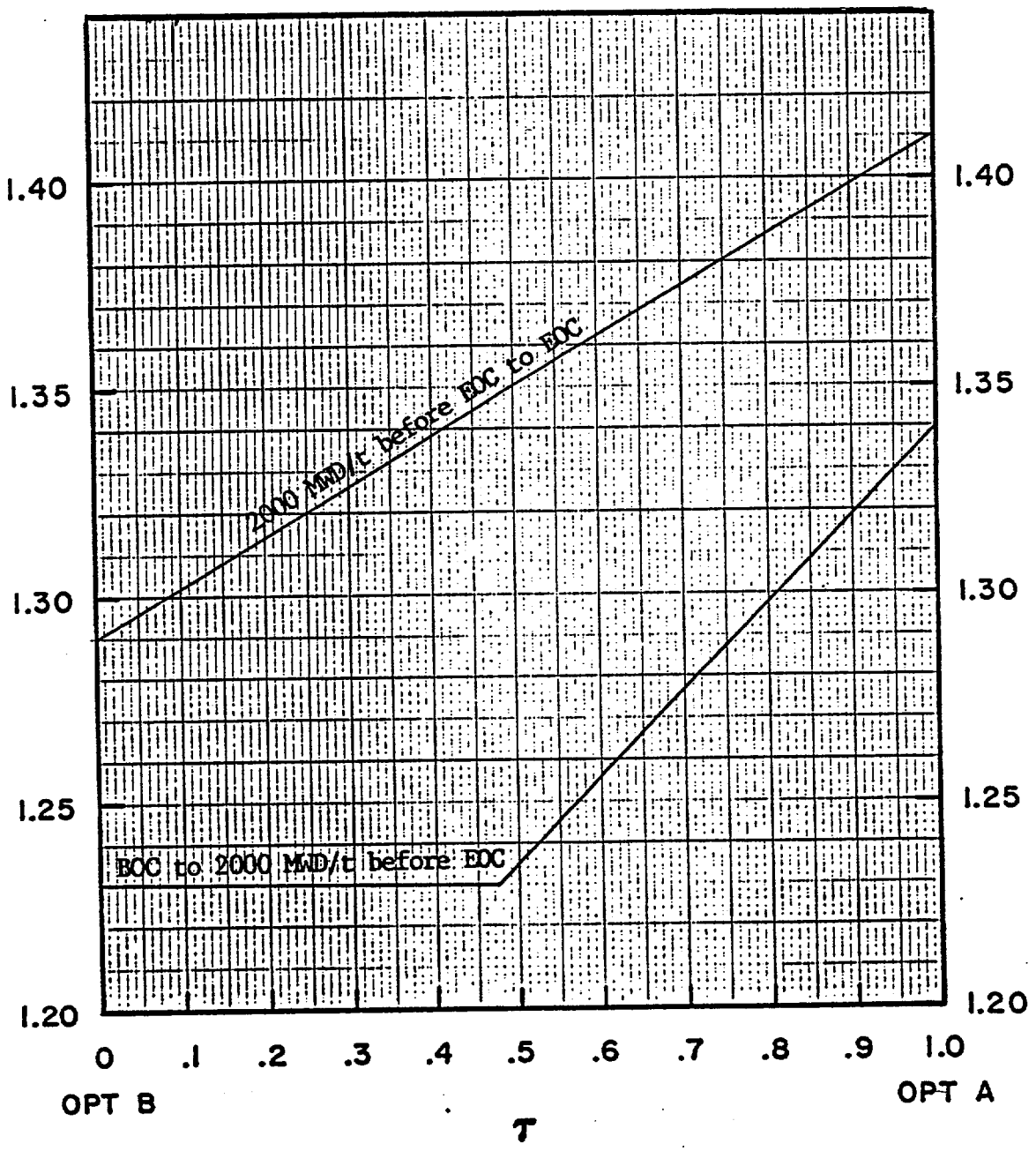
** These values shall be increased by 0.01 for single loop operation.

*** Applicable to all P8X8R fuel bundles including BP8X8R and the P8DRB285 (Reload 5) types.

PEACH BOTTOM UNIT 2

FIGURE 3.5.K.2 MCPR OPERATING LIMIT vs $\tau$

FUEL TYPE P8X8R*




* Applicable to all P8X8R fuel bundles including BP8X8R and the P8DRB285 (Reload 5) Types

Amendment No. 78, 90, 70, 86, 108

PEACH BOTTOM UNIT 2

FUEL TYPE P8DRB299

BP8X8R Fuel Type BP8DRB299

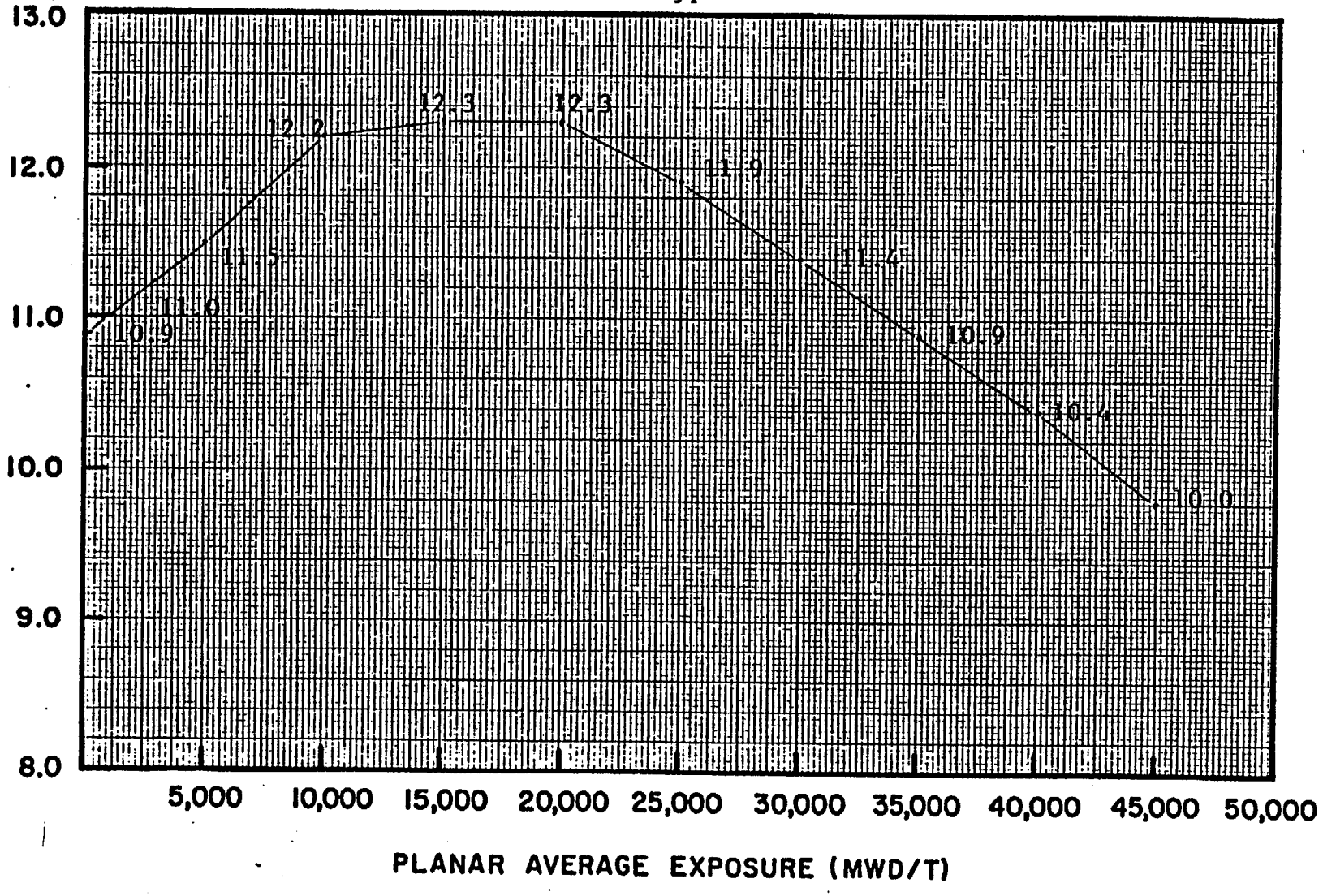



FIGURE 3.5.1J MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE VERSUS PLANAR AVERAGE EXPOSURE

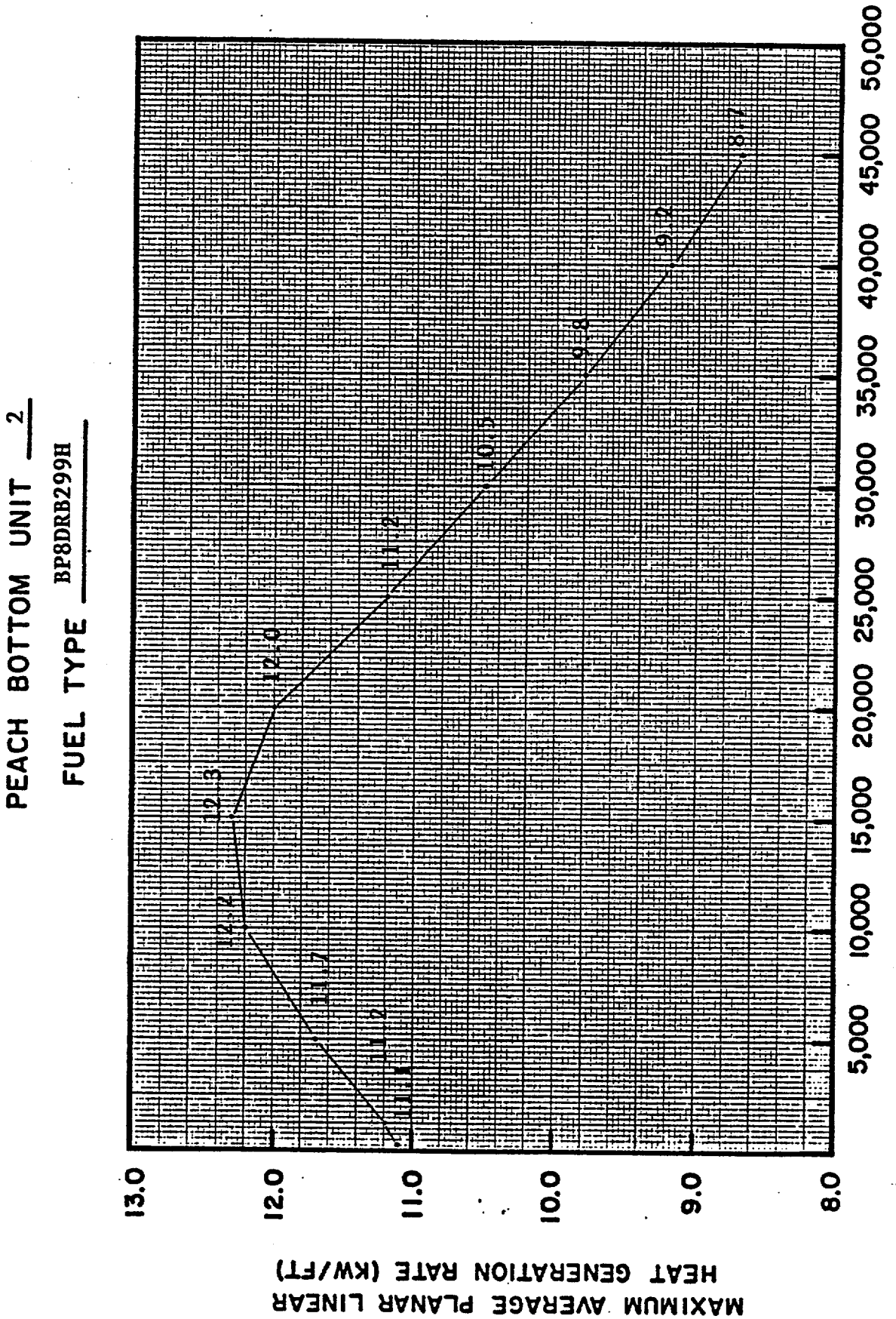



FIGURE 3.5.1.1 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE VERSUS PLANAR AVERAGE EXPOSURE

Amendment No. 108

## 5.0 MAJOR DESIGN FEATURES

### 5.1 SITE FEATURES

The site is located partly in Peach Bottom Township, York County, partly in Drumore Township, Lancaster County, and partly in Fulton Township, Lancaster County, in southeastern Pennsylvania on the westerly shore of Conowingo Pond at the mouth of Rock Run Creek. It is about 38 miles north-northeast of Baltimore, Maryland, and 63 miles west-southwest of Philadelphia, Pennsylvania. Figures 2.2.1 through 2.2.4 of the FSAR show the site location with respect to surrounding communities.

### 5.2 REACTOR

A. The core shall consist of not more than 764 fuel assemblies.

B. The reactor core shall contain 185 cruciform-shaped control rods.

### 5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2.2 of the FSAR. The applicable design codes shall be as described in Table 4.2.1 of the FSAR.

### 5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2.1 of the FSAR. The applicable design codes shall be as described in Appendix M of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.3.4 of the FSAR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-44

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. 2

DOCKET NO. 50-277

## 1.0 INTRODUCTION

By letter dated September 7, 1984, Philadelphia Electric Company (the licensee) made application to amend the Technical Specifications of Peach Bottom Atomic Power Station, Unit 2, to permit reloading and operation of the unit for Cycle 7. In support of this application the licensee submitted a reload report (Reference 1), an update of the Loss of Coolant Accident (LOCA) analysis (Reference 2) and a single loop operation report (Reference 3).

### 1.1 Description of the Proposed Amendment Changes Relating to the Cycle 7 Core

The proposed amendment to the Peach Bottom Unit 2 Technical Specifications would:

1) Modify the bases of the Standby Liquid Control System to specify the required core boron concentration rather than the required shutdown margin,

2) Change the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) reduction factor to be applied during single loop operation,

3) Revise the Minimum Critical Power Ratio (MCPR) limits for Cycle 7 operation,

4) Provide MAPLHGR limits for the two new fuel types inserted for Cycle 7, and

5) Revise the Design Features section of the Technical Specifications to permit introduction of the improved Hybrid I control rods.

Each of these changes to the Technical Specifications is discussed in Section 2.5 below.

## 2.0 EVALUATION

### 2.1 Fuel Mechanical Design

The fuel to be inserted into the core for Cycle 7 is similar to that customarily used for BWR reloads and is described in Reference 4. This report has been approved by the NRC staff (Reference 5), and we conclude that no further review of the fuel mechanical design is required.

### 2.2 Nuclear Design

The nuclear design and analysis of the Cycle 7 reload was performed with methods and techniques which are described in Reference 4 and which are used in all reload analyses performed by General Electric. The results of the analyses are within the range of those customarily found for reload cores and are acceptable. We conclude that the nuclear design and analysis of the Cycle 7 reload is acceptable.

### 2.3 Thermal-Hydraulic Design

The methods and procedures employed in the thermal-hydraulic design and analysis of the Cycle 7 core are described in Reference 4. The value of 1.07 for the safety limit MCPR, approved in that reference, is used for Cycle 7. The methods and procedures used to obtain the operating limit MCPR were those described in Reference 4 and are acceptable.

Thermal-hydraulic stability for BWRs is presently the subject of a generic study and the General Electric design methods for prediction of core stability are under review. Our review of the design methods using FABLE has progressed sufficiently that we have assigned a 20 percent uncertainty to the calculated decay ratio. Thus, we expect that Peach Bottom 2 Cycle 7, which has a calculated core stability decay ratio of 0.87, may be unstable under certain abnormal, but possible, operating conditions in the low flow-high power region of the operating map. However, we have also concluded that the core stability characteristics are essentially unchanged from the previous cycle, which had a calculated decay ratio of 0.85. Therefore, any corrective measures required upon completion of our generic study are unrelated to this reload and may be implemented separately. In the interim, we conclude that there is reasonable assurance that continued operation of Peach Bottom 2 will not result in power oscillations leading to violation of specified acceptable fuel design limits (SAFDL) for the reasons that follow:

1) Peach Bottom 2 and other reactors with comparable core designs have many years of operating history without known incidents of power oscillations which resulted in exceeding the SAFDL.

2) Philadelphia Electric Company is aware of the operating recommendations provided in the General Electric Service Information Letter (SIL-380) to avoid operating regions of potential instability and to detect and suppress power oscillations if they should occur.

We conclude that the thermal-hydraulic design and analysis of the Peach Bottom 2 Cycle 7 core is acceptable.

## 2.4 Transient and Accident Analyses

The transient and accident analyses for Cycle 7 have been performed with the methods described in Reference 4 and are reported in Reference 1. The limiting non-pressurization event is the Rod Withdrawal Error resulting in a required operating limit MCPR (OLMCPR) of 1.23. The limiting pressurization event for option A is the Load Rejection without Bypass resulting in a required OLMCPR of 1.30 during the early part of the cycle and 1.39 at the end of cycle. For option B the limiting event is Feedwater Controller Failure (OLMCPR = 1.15) during the early part of the cycle and is Load Rejection without Bypass (OLMCPR = 1.27) at end of cycle.

The LOCA has been reanalyzed to obtain MAPLHGR curves for the new fuel assembly types to be inserted for Cycle 7. A cycle specific rod drop accident analysis has been performed for Cycle 7 resulting in a peak fuel enthalpy of 241 calories per gram. This meets our acceptance criterion of 280 calories per gram and is acceptable. Because the transient and accident analyses have been performed by previously approved methods and the results meet our acceptance criteria, we conclude that they are acceptable.

## 2.5 Technical Specifications

### 2.5.1 Basis for Standby Liquid Control System

The basis for meeting the boron concentration and volume limits in the Standby Liquid Control System has been altered to require the system to be capable of inserting boron to a given concentration in the core within a given time rather than to provide a fixed shutdown margin. Cycle specific calculations are then performed to determine the shutdown margin obtained. The revised procedure is more straightforward and is common practice in BWR reloads. We find it acceptable for Peach Bottom Unit 2.

### 2.5.2 MAPLHGR Reduction Factor

When operating with a single loop, it is necessary to reduce the OLMAPLHGR values in order to maintain the margin to peak clad temperature limits in the LOCA analysis. The reduction factor has been calculated by methods described in Reference 3, and the Technical Specification value is consistent with the results in that reference. It is therefore acceptable.

### 2.5.3 OLMCPR Values

The proposed Technical Specification values of the OLMCPR are conservative with respect to the values reported in Reference 1 and are acceptable.

### 2.5.4 MAPLHGR Limits for New Fuel

The MAPLHGR limits in the Technical Specifications are consistent with those given in Reference 2 and are acceptable.

2.5.5 Hybrid I Control Rods

Use of Hybrid I control rods in BWRs has been reviewed by the NRC staff and found to be acceptable. Their use in Peach Bottom Unit 2 is therefore acceptable. The description of control rods is being deleted from the Technical Specifications. Since the standard control rods are described in the FSAR and the Hybrid I rods are described in approved Topical Report NEDE-22290-A, we find this to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

4.0 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.

Dated: March 19, 1985

The following NRC personnel have contributed to this Safety Evaluation: W. Brooks and G. Schwenk.

References


1. "Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit 2, Reload 6", General Electric, Report 22A8597, June, 1984.

2. Errata and Addenda Sheet No. 10 to NEDO-24081, "LOCA Analysis for Peach Bottom APS Unit 2," June, 1984.

3. Errata and Addenda Sheet No. 2 to NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3 Single Loop Operation," June, 1984.

4. GESTAR II - "General Electric Standard Application for Reactor Fuel" - NEDE-24011-P-A-6, April, 1983.

5. GESTAR II - "General Electric Standard Application for Reactor Fuel" - NEDE-24011-P-A-6, April, 1983. Appendix C.