May 4, 1983

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See Correction letter of 5/19/83

Docket No. 50-278

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

Dear Mr. Bauer:

The Commission has issued the enclosed Amendment No. 92 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Unit No. 3. The amendment revises the Technical Specifications (TSs) in response to your application dated December 30, 1982, as supplemented April 6, 1983.

The changes to the TSs permit reactor operation of Peach Bottom Unit No. 3 with the Reload Number 5 core (Cycle 6).

Copies of our Safety Evaluation and a related Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Gerald E. Gears, Project Manager
Operating Reactors Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 92 to DPR-56
2. Safety Evaluation
3. Notice of Issuance

8305180007 830504
PDR ADOCK 05000278
P PDR

Noted objection to form of amendment & FR notice. SER review not requested

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD		
SURNAME	RIngram	GGears;cf	JStolz	GLainas	CUTCHIN		
DATE	5/2/83	4/28/83	5/2/83	5/2/83	5/3/83		

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

May 4, 1983

DISTRIBUTION
Docket File
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RIngram

Docket No. 50-278

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Amendment No. 92.
  Referenced documents have been provided PDR.

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE →	ORB#4:DL					
SURNAME →	RIngram;cf					
DATE →	5/5/83					

NRC FORM 102 (1-76)

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

May 4, 1983

Docket No. 50-278

Mr. Edward G. Bauer, Jr.
Vice President and General Counsel
Philadelphia Electric Company
2301 Market Street
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Gerald E. Gears

Gerald E. Gears, Project Manager
Operating Reactors Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 92 to DPR-56
2. Safety Evaluation
3. Notice of Issuance

Philadelphia Electric Company

cc w/enclosure(s):

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6th and Walnut Streets
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Generation Division - Nuclear
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Regional Administrator
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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. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
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 December 30, 1982, as supplemented April 6, 1983, 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. 92, 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

John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 4, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 92

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

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

Remove Pages	Insert Pages
iv	iv
119	119
133a	133a
133d	133d
133e	133e
142	142
142a	142a
142b	-
142c	-
142e	142e
142f	-
-	142j
241	241

PBAPS

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 increasing testing of the operable redundant component.

LIMITING CONDITIONS FOR OPERATION

3.5.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 operation. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the following reduction factors: 0.71 for 7X7 fuel; 0.83 for 8X8 fuel; 0.81 for PTA, 8X8R, P8X8R, and LTA 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.

3.5.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.

$LHGR \leq LHGRd$

LHGRd = Design LHGR
13.4 kW/ft for all 8X8 fuel

SURVEILLANCE REQUIREMENTS

4.5.I Average Planar LHGR

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

4.5.J Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power.

Amendment No. 33, 47, 62, 77, 79, 92

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
PTA &P 8X8R	1.26	1.27
LTA	1.26	1.28

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

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

Amendment No. ~~42~~, ~~62~~, 77, ~~79~~, ~~85~~, 92

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
PTA &P 8X8R	1.33	1.39
LTA	1.33	1.40

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

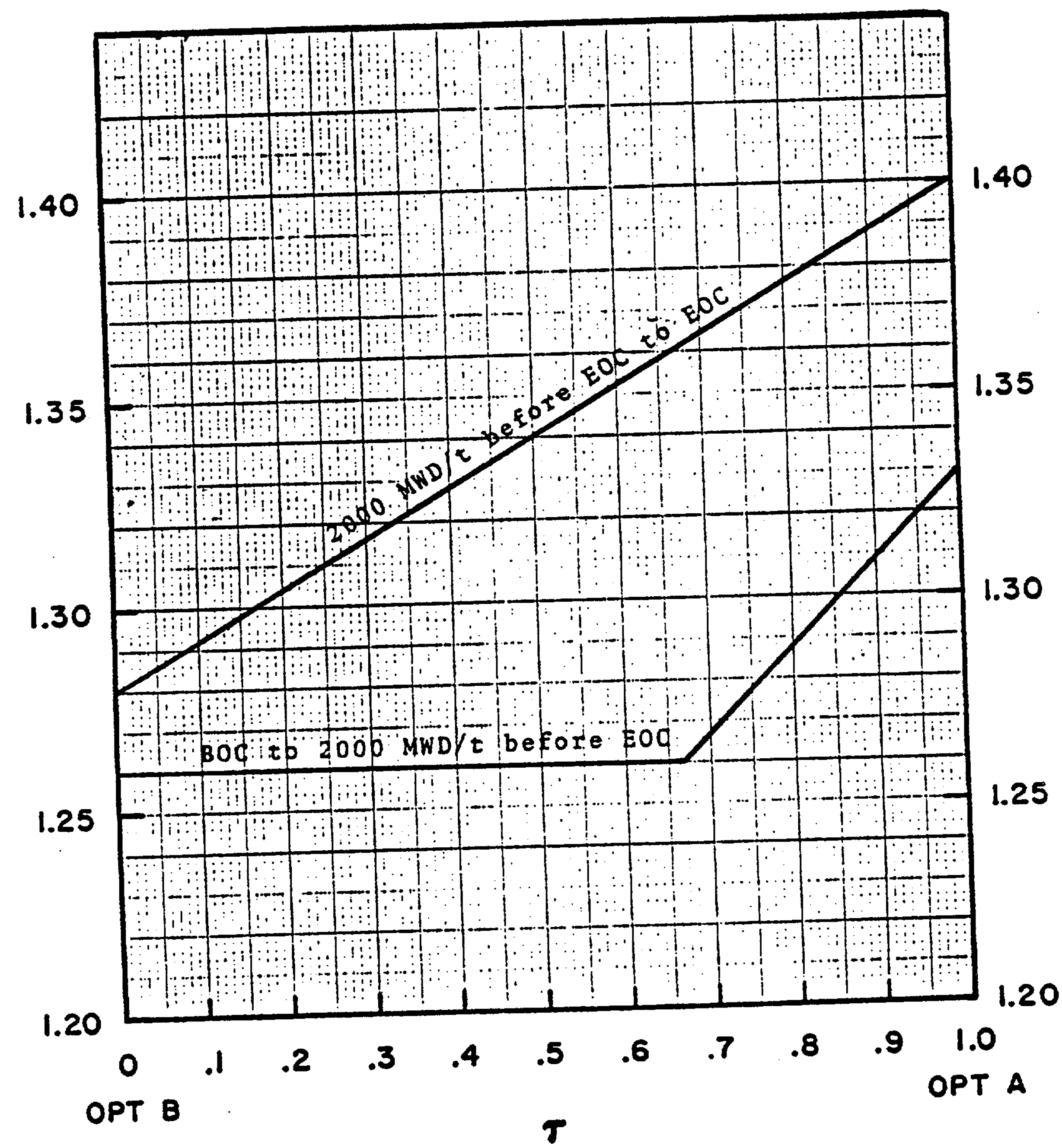
** These values shall be increased by 0.01 for single loop operation.

Amendment No. ~~79~~, ~~85~~, 92

PEACH BOTTOM UNIT 3

FIGURE 3.5.K MCPR OPERATING LIMIT vs τ

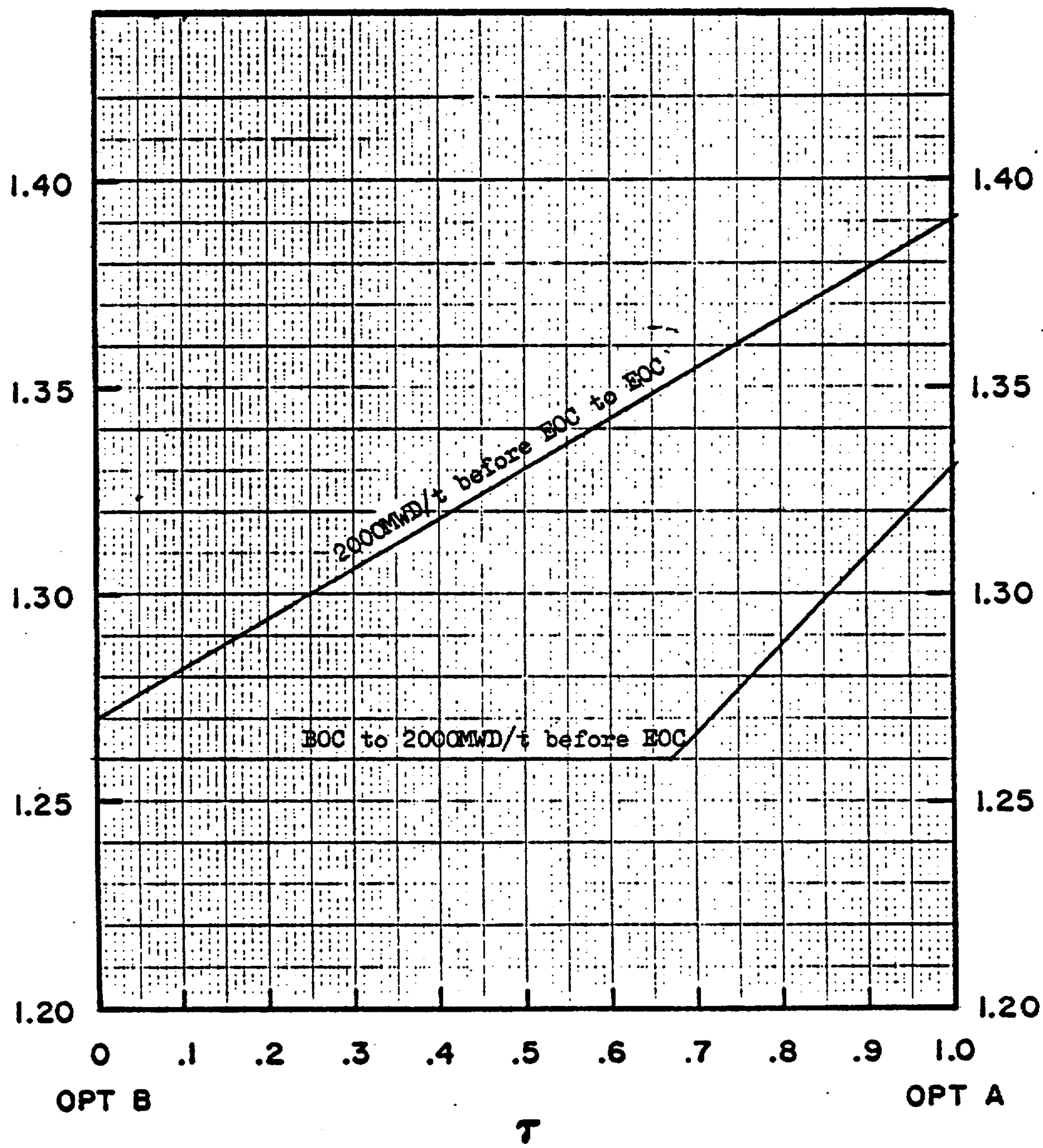
FUEL TYPE LTA



Amendment No. ~~47~~, ~~79~~, ~~85~~, 92

FIGURE 3.5.K.2 MCPR OPERATING LIMIT vs τ

FUEL TYPE PTA & P8X8R



Amendment No. ~~47~~, ~~79~~, ~~85~~, 92

PEACH BOTTOM UNIT 3

FUEL TYPE P8X8Q LTA

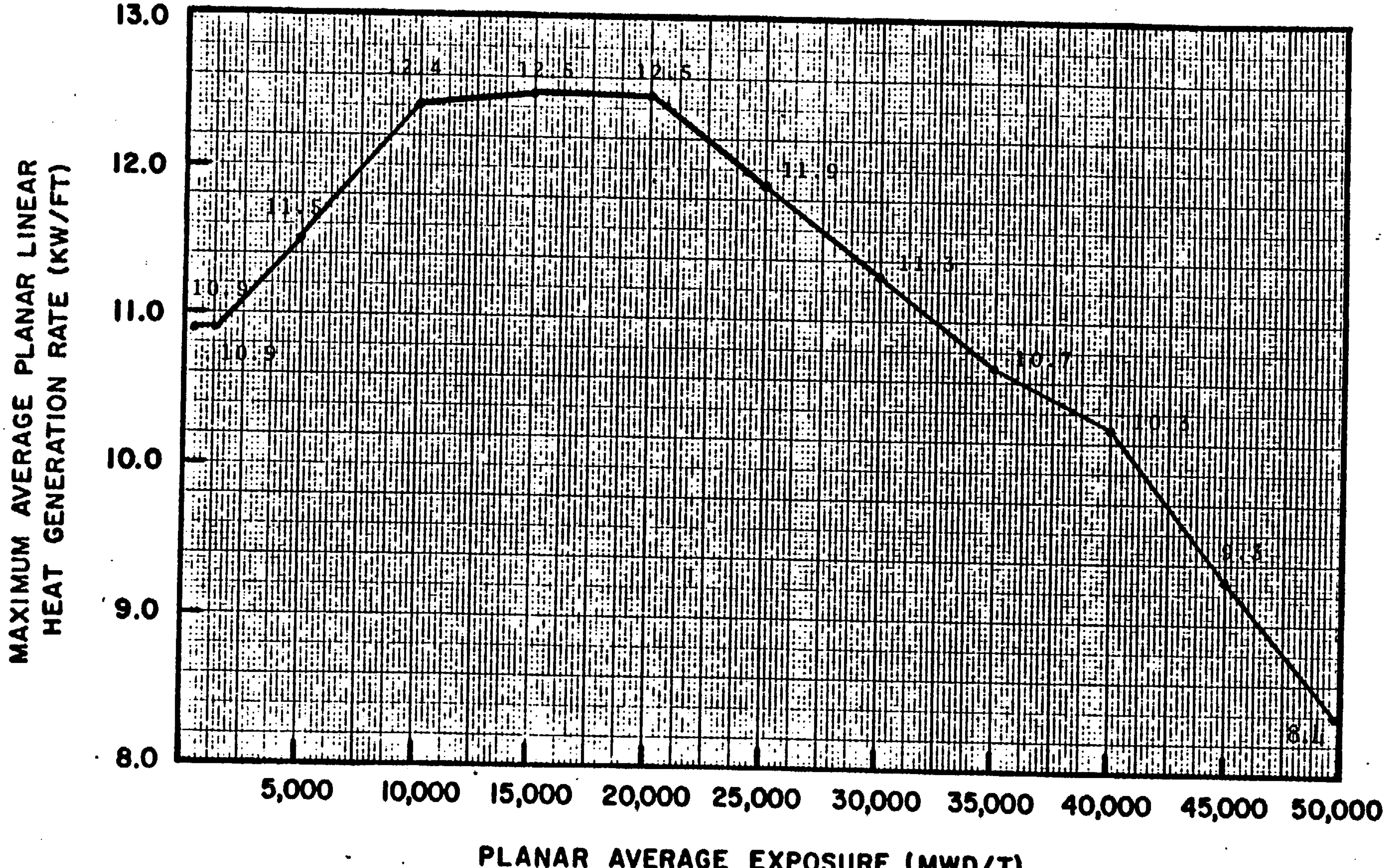


FIGURE 3.5.1.K MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE VERSUS PLANAR AVERAGE EXPOSURE

PBAPS

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. One Pressurized Test Assembly and four Lead Test Assemblies may be inserted in the core.

C. 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. 92 TO FACILITY 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 3

DOCKET NO. 50-278

1.0 Introduction

The Philadelphia Electric Company (PECo or the licensee) requested (Ref. 1) that the Technical Specifications (TSs) appended to Facility Operating License DPR-56 for Peach Bottom Atomic Power Station Unit 3 be amended to accommodate the fifth refueling of the reactor. Specifically, the requested TS changes were intended to accomplish the following:

1. Identify the operating limits for all fuel types for Cycle 6 operations.
2. Permit continued operation of a Pressurized Test Assembly (PTA) after reconstitution.
3. Permit operation with four new Lead Test Assemblies (LTAs).
4. Incorporate Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the new LTAs and extended exposure MAPLHGR limits for the PTA.

5. Permit operation with up to six General Electric (GE) hafnium Hybrid I Control Rods (HICRs).
6. Modify bases to delete reference to a specific shutdown margin value provided by the Standby Liquid Control System.

An analysis of the safety considerations involved in the reactor refueling and the Cycle 6 operating limits for all fuel types is set forth in Reference 2, which was filed along with other documents (Refs. 3,4) in December 1982. Other information (Refs. 5-10) relevant to the Cycle 6 reload had been provided earlier.

2.0 Fuel System Design

2.1 Background

The Peach Bottom 3 Cycle 6 core will contain 764 fuel assemblies of which 284 will be fresh reload 5 assemblies. The core composition is summarized in Table I. Detailed descriptions of the four LTAs, the one PTA, and the 759 standard fuel assemblies are provided in References 6, 7, and 8, respectively. Since the standard fuel assemblies are comprised of a reviewed and approved design, this safety evaluation mainly addresses the four LTAs and the PTA, along with six GE HICRs described in Reference 3. The fuel system design aspects of the six TS change objectives listed in Section 1.0 of this safety evaluation are addressed in the following subsections.

TABLE I

PEACH BOTTOM UNIT 3 CYCLE 6 FUEL BUNDLES

	Fuel Type	Cycle Loaded	Number
Irradiated	P8DRB284H	4	263
	P8DRB299	5	216
	PTA	2	1
New	P8DRB284H	6	56
	P8DRB299	6	224
	PBLTA 1	6	2
	PBLTA 2	6	2
Total			764

2.2 Operating Limits for Cycle 6 Fuel Types

Information related to fuel system operating limits is contained in Reference 2 and the related TS changes were submitted with Reference 1. Reference 2 contains analytical results of the safety considerations involved in the reactor refueling and Cycle 6 operating limits. Thus, core-wide critical power ratio changes (ΔCPRs) for several transients, including load rejection without bypass, loss of feedwater heating, feedwater controller failure, and rod withdrawal errors, are provided for the PTA and LTAs as functions of various input parameter assumptions. Minimum critical power ratios (MCPRs) are listed in that report along with maximum linear heat generation rates (LHGRs) for the rod withdrawal error and misoriented bundle events. As discussed in Section 4.0, Thermal and Hydraulic Design, of this safety evaluation, the proposed operating limits and corresponding TSs were reviewed and found acceptable.

2.3 Operation with Reconstituted Pressurized Test Assembly

A PTA, described in Reference 7, was originally inserted in the Peach Bottom 3 reactor during the Spring 1977, Reload 1 refueling. The purpose of continued PTA operation is to obtain fission gas measurements from the PTA in conjunction with an extended exposure program. Twenty-two fuel rods will be removed from the PTA and replaced with irradiated rods from an 8DRB283 bundle (initially inserted as part of Reload 2), which is due to be discharged at end-of-cycle(EOC)-5 (Ref. 17). The estimated PTA bundle average exposure at the Spring 1983 outage is approximately 30 GWd/MT. An additional cycle of operation would extend the fuel bundle average exposure to approximately 35 GWd/MT with a peak pellet exposure approaching 46 GWd/MT.

An analysis of the safety considerations involved in continuing the use of the PTA is set forth in Appendix C of Reference 2. As indicated therein, reconstitution of the PTA could result in a slight increase in peak cladding temperature (PCT) due to stored energy and local power distribution effects. The resulting increase in PCT on account of these effects is insignificant (10 to 20°F) (Ref. 2) compared to the margin to the PCT loss-of-coolant-accident (LOCA) limit of 2200°F (the magnitude of the maximum PCT of the non-reconstituted PTA is 1923°F). Since the enrichment of the replacement rods was selected to assure that the reactivity of the reconstituted PTA will not exceed that of the non-reconstituted PTA, since the peak linear heat generation rate of the reconstituted PTA is well within the operating limit of 13.4 kW/ft, and since the LOCA limits are not exceeded, we conclude that continued operation of the PTA during Cycle 6 is acceptable.

2.4 Lead Test Assembly Operation

The TS changes requested by PECo in Reference 1 would permit operation with four LTAs of fuel type P8DQB326 (for fuel description, see GESTAR-II, Ref. 8). Two of the LTAs will utilize an improved pressure drop spacer (low ΔP spacer), while the other two LTAs will have the normal spacer provided for 8x8R fuel. The LTAs will also incorporate several other features similar to those submitted for Browns Ferry 3 in the Fall of 1981. Analyses of the safety considerations involved with the LTA program are provided in Refs. 1 and 6. The proposed TS changes incorporate MAPLHGR limits for the four new LTAs and extended exposure MAPLHGR limits for the one PTA. Our review of those subjects is described in the following subsection (2.5) of this safety evaluation.

With regard to LTA unique inputs and analyses (described in Attachment 2 of Reference 6 and in Reference 20), both core-wide and localized transients and accidents were considered. The LTAs were stated (Ref. 6) to have been analyzed using GESTAR-II (Ref. 8) methods and to have met all applicable GESTAR-II approved criteria. Except for the rotated bundle event, the calculated MCPRs did not violate the safety limit MCPR. For that event, however, special loading surveillance should mitigate against the possibility of a misoriented bundle.

Since the number of LTAs (4) is small, since they have been designed and analyzed using approved methods, and since, except for the rotated bundle event, no design or operating limits will be exceeded, we conclude that there is reasonable assurance that the insertion and operation of the four LTAs will not pose an unacceptable risk to the public health and safety.

We expect to be informed in a timely manner concerning the results of the measurements to be conducted on the LTAs. As indicated in Reference 6, those measurements, as currently envisioned, are to consist of overall bundle visual examinations, bundle and rod length measurements, rod integrity and profilometry measurements, corrosion thickness measurements, fission gas sampling, spacer spring relaxation and possibly gamma scans. It should be noted that, while PECo has stated (Ref. 6) that GE will summarize the results from the LTA program in GE's fuel experience reports "in a timely manner," those reports have had about a five-year periodicity. Thus, in the interest of timeliness, we will expect PECo to provide an informal summary of the LTA examinations within six months following each refueling outage during their lifetime in reactors.

2.5 MAPLHGR Limits

Analyses of the safety considerations involved in the proposed MAPLHGR limits for the four LTAs and extended exposure limits for the one PTA are provided in References 4 and 5. Although the methodology used is generally applicable for these limits, we believe that the effects of enhanced fission gas release in high burnup fuel (above 20 MWd/kgU) were not adequately considered in the generic analysis. In response to this concern, GE requested (Refs. 9 and 10) that credit for approved, but unapplied, emergency core cooling system (ECCS) evaluation model changes, and calculated PCT margin, be used to avoid MAPLHGR penalties at higher burnup. This proposal was found acceptable (Ref. 11) provided that certain plant-specific conditions were met. PECo has stated (Ref. 12) that the GE proposal is applicable to both Peach Bottom Units 2 and 3. On the basis of this finding, we conclude that the MAPLHGR limits proposed for Peach Bottom 3 Cycle 6 are acceptable.

2.6 Operation with Hafnium (GE Hybrid I) Control Rods

The TS changes requested by PECo would permit operation with up to six Type II (surveillance version) GE HICRs. The HICR Type II test program is designed to provide pre-commercial test data for GE's new Type I (production version) HICR. An analysis of the safety considerations involved in the HICR Type II test program is set forth in Reference 3 (NEDE-22290), which is a generic report describing the design and the analyses performed by GE to demonstrate the safety of both the Type I and Type II hafnium-hybrid control rods.

The principal objectives of the HICR are to (1) increase control rod assembly life and (2) eliminate cracking of absorber tubes containing boron carbide (B_4C). The major design changes that are intended to ensure that those objectives are met are (1) the use of an improved B_4C absorber rod tube material to eliminate stress corrosion cracking during the lifetime of the assembly and (2) replacement of some B_4C absorber rods with solid hafnium absorber rods. In addition, there are other material and dimensional changes, including a reduction in sheath wall thickness and a change in the pin and roller materials from Stellite to other materials discussed in Reference 13. Other variables included the location of the hafnium rods, the type of tubing used for B_4C rods and the use of clad versus unclad Hf.

Due to the complexity of the HICR test program (as evidenced by the large number of variables to be examined), a meeting (Ref. 4) was held with GE and PECo to discuss the program in Peach Bottom 3 as well as the overall R&D program, analyses, surveillance, etc. performed or underway by GE in support of the HICR design. The purpose of the meeting was actually two-fold:

1. To support the proposed amendment to the Peach Bottom 3 operating license to permit HICR use.
2. To support the generic use of HICRs in BWRs.

Because the generic review is much broader in scope than could be accommodated by the tight schedule required for Peach Bottom 3 Reload 5, this safety evaluation addresses only the issues involving the six surveillance HICRs. The results of the generic review will be reported separately as a safety evaluation of the GE topical report, NEDE-22290 (Ref. 3).

With regard to the Peach Bottom 3 Cycle 6 use of the six Type II HICRs, the key issues concerned the potential effects of the changes in component materials and dimensions. The safety considerations involved are discussed below for each design change.

Pins and Rollers - As indicated in EPRI NP-2329 (Ref. 13), the pin and roller materials currently in use in BWRs are cobalt-base alloys (Haynes 25 and Stellite 3, respectively). Because cobalt-60 is an isotope that contributes significantly to plant radiation buildup, there is an incentive to replace the cobalt alloys with non-cobalt alloys and thus reduce personnel radiation exposure during plant maintenance. EPRI NR-2329 describes an extensive program at GE to qualify substitute non-cobalt alloy control rod pin and roller materials. Wear resistance measurements in a simulated BWR environment (excluding irradiation), coupled with impact strength and corrosion tests, indicate that the non-cobalt alloys have equivalent or better wear resistance, superior impact strength and similar corrosion resistance to the conventional cobalt alloys. Though the effects of irradiation were not investigated in those tests, reactor tests have been initiated at a control cell BWR and at a conventional core BWR. We conclude that the substitution of the non-cobalt alloys for Haynes 25 and Stellite 3 pins and rollers in the six Type II surveillance HICRs is acceptable, based on the results of the tests described in EPRI NP-2329 and our expectation that (a) the surveillance described on page S-5 of EPRI NP-2329 will be carried out, (b) the results of that surveillance will be reported in a timely fashion, and (c) surveillance of the six HICRs in Peach Bottom 3 will also be conducted and reported.

Control Rod Tubing Material - As indicated on page 2-2 of NEDE-22290 (Ref. 3), the B_4C absorber rod tubing for the Type I (production version) control rods is a high purity Type 304 stainless steel, while the Type II (development) control rods will also contain some high purity Inconel 600 as an alternate absorber tube material. Both of these alloys have undergone extensive qualification testing and evaluation including laboratory testing, correlation of field performance with intergrannular stress corrosion cracking susceptibility tests, and assessment of archival materials. In addition, an extensive surveillance program, including visual examinations, dimensional measurements, eddy current testing, neutron radiography, isotopic determinations, and steam corrosion testing (see p. 5-10 of Ref. 3) is planned. Based upon the information provided in Ref. 3 and in the meeting described in Ref. 9, we conclude that the use of the new absorber tube alloys is acceptable for the six Type II HICRs. We expect to be informed of the results from the HICR surveillance program on the absorber tube materials as those results relate to the potential performance of the production version HICRs.

Absorber Material - As indicated in NEDE-22290 (Ref. 3), three of the B_4C absorber rods per blade (12 in each control rod assembly) in the present BWR 2-4 D lattice CRA design will be replaced with solid Hafnium rods. In the Type I production version HICRs, the Hf rods are unclad and located at the tip positions of each blade. The three main concerns related to the use of Hf rods involve (a) the increase in weight, (b) the thermal expansion of Hf relative to the absorber cladding material, and (c) the corrosion resistance of unclad Hf.

With regard to the increased weight resulting from the higher density of Hf relative to the B_4C it replaces, the reduction in blade sheath thickness (and weight) compensates for the increase in absorber material weight. The resultant sheath thickness falls within the range of GE design experience, and the increased fuel channel clearance should reduce potential fuel channel interference. From a mechanical design standpoint, therefore, there is reasonable assurance that the design changes related to the increased weight of the absorber material have been adequately accounted for in the six Type II HICRs. The planned surveillance of the HICRs should provide confirmation of this.

With regard to the thermal expansion and irradiation growth considerations, the coefficient of thermal expansion of Hf is approximately half that of Type 304 stainless steel and Inconel 600 (the B_4C absorber tubing materials), and is comparable to an alternate cladding material used for some of the Hf rods in the Type II HICRs. Inasmuch as only a few Type II rods will have the alternate cladding material, any adverse effects, which are not anticipated, should not be significant. The irradiation growth of hafnium is expected to be small. Bare hafnium absorber rods in the Peach Bottom 2 reactor have shown virtually no change in length or diameter after 18 months service. Since dimensional measurements will be made of the Hf rods at 18-24 month intervals as part of Type II HICR surveillance program, the irradiation growth will readily be monitored.

With regard to the corrosion of hafnium in a BWR environment, there is significantly more information regarding PWR use of hafnium (because of naval reactor use). GE did present some data (Refs. 3 and 4), however. Those data showed that the corrosion behavior of hafnium in high temperature water and steam is superior to that of Zircaloy-2. In addition, an experimental, bare Hf control rod in Peach Bottom 2 has shown little corrosion after 1.5 years exposure (Refs. 14 and 15). The planned Type II HICR surveillance program is intended to include metallographic examinations of the Hf rod hydriding behavior and corrosion characteristics. We conclude, therefore, that the corrosion behavior of the Type II HICR Hf rods has been adequately addressed for Peach Bottom 3 Cycle 6 operation.

2.7 Fuel System Design Conclusions

We have reviewed the information submitted on the Cycle 6 operation of Peach Bottom 3, including the design, analysis, testing, and proposed surveillance of a PTA, four LTAs, and six Type II HICRs. We find the Peach Bottom 3, reload 5 proposed refueling and related TS changes acceptable from a mechanical design standpoint.

3.0 Nuclear Design

The nuclear design of the proposed reload was performed by the approved methods of Reference 8 including that of the LTAs. The nuclear parameters for the reload are within the range of those normally seen for BWR reloads and are acceptable.

4.0 Thermal and Hydraulic Design

The objective of the thermal-hydraulic review is to confirm that (a) the thermal-hydraulic design of the core has been accomplished using acceptable methods, (b) the design provides an acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients, and (c) the design is not susceptible to thermal-hydraulic instability.

The thermal-hydraulic review includes the following areas: (1) safety limit MCPR, (2) operating limit MCPR, (3) thermal-hydraulic stability, and (4) changes to Tables 3.5.K.2 and 3.5.K.3 and Figures 3.5.K.1 and 3.5.K.2 of the TSs.

The licensee has submitted the analysis report for Cycle 6 operation at rated core flow conditions (Ref. 2). Discussion of our review concerning the thermal-hydraulic design for Cycle 6 operation follows:

4.1 Safety Limit MCPR

The safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during normal and anticipated operational transients. As stated in Reference 8, the safety limit MCPR is 1.07. The safety limit MCPR of 1.07 is used for Peach Bottom 3 Cycle 6 operation.

4.2 Operating Limit MPCR

The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR). The ΔCPR values given in Section 9 of Reference 2 are plant specific values calculated by including the ODYN Methods. The calculated ΔCPRs are adjusted to reflect either Option A or Option B ΔCPRs by employing the conversion methods described in Reference 16. The MCPR values are determined by adding the adjusted ΔCPRs to the safety limit MCPR. Section 11 of Reference 2 presents both the cycle MCPR values for the pressurization and non-pressurization transients. The maximum cycle MCPR values (Options A and B) in Section 11 are specified as the operating limit MCPRs and incorporated into the TSs. Since the approved method was used to determine the operating limit MCPRs to avoid violation of the safety limit MCPR in the event of any anticipated transients, we conclude that these limits are acceptable.

4.3 Thermal-Hydraulic Stability

The results of thermal-hydraulic analysis (Ref. 2) show that maximum reactor core stability decay ratio is about 0.98, which is comparable to the calculated value for Peach Bottom 2 Reload 3, which has been previously approved. Since operation in the natural circulation mode is prohibited by TS 2.1.A.4, there will be added margin to the stability limit. We therefore conclude that the thermal-hydraulic stability results are acceptable for Cycle 6 operation.

4.4 Changes to the Technical Specifications

Figures 3.5.K.1, 3.5.K.2 and Tables 3.5.K.2, 3.5.K.3 of the TSs have been modified to include the operating limit MCPRs for Cycle 6 operation. Using Option A, the operating limit MCPRs would be 1.33 for Cycle 6 fuels at burnup conditions from BOC to 2000 MWD/t before EOC, and 1.39 for PTA, P8X8R fuel types and 1.40 for LTA at burnup conditions from 2000 MWD/t before EOC to EOC. Using Option B, the operating limit MCPRs would be 1.26 for Cycle 6 fuels at burnup conditions from BOC to 2000 MWD/t before EOC, and 1.27 for PTA, P8X8R fuel types and 1.28 for LTA at burnup conditions from 2000 MWD/t before EOC to EOC.

4.5 Thermal and Hydraulic Design Evaluation Summary

We find that approved thermal-hydraulic methods have been used and that results of analyses support the proposed limit MCPRs, which avoid violation of the safety limit MCPR for design transients. We conclude that this core reload will not adversely affect the capability to operate Peach Bottom 3 safely during Cycle 6 operation and that the revised Figures 3.5.K.1, 3.5.K.2 and Tables 3.5.K.2, 3.5.K.3 of the TSs discussed above are acceptable.

5.0 Transients and Accidents

As described in Section 2.4 above, the analyses of the transients and accidents have been performed with the approved methods of Reference 8, and with the exception of the fuel misorientation event, meet all acceptance criteria. The fuel misorientation event is discussed below.

The effect of the presence of the HICRs on the results of these events is expected to be negligible for the following reasons:

1. Only six of the HICRs are present,
2. The nuclear characteristics of the hybrid rods resemble closely those for standard rods, and
3. The scram speeds are identical to the standard rods.

5.1 Fuel Assembly Misorientation for Lead Test Assembly

When analyzed with standard procedures (NEDO-24011-A-US, Section S.2.5.4.2), the misorientation of one type of LTA (PBLTA1) can lead to a MCPR value of 1.06 when the core is operated at the proposed operating limit. The licensee states that the proposed operating limit MCPR need not be altered to accommodate this event since special precautions will be taken to prevent it. We find this position to be acceptable for the following reasons:

1. There are only four LTAs - only two of which are a concern for this event.
2. The licensee proposed to initiate special procedures for the LTAs during this cycle to prevent misorientation.
3. The calculation of MCPR for this event tends to be conservative and the variation from the safety limit is small.

6.0 Technical Specification Change

PECo wishes to delete the reference to a five percent shutdown margin in the bases to the TSs for the Standby Liquid Control System (Specification 3.4). This would bring the specification into correspondence with that in the Standard Technical Specifications (for BWRs, Specification 3/4.1.5). The actual value of the shutdown margin is provided for each cycle as part of the supplemental reload licensing submittal for the 660 parts per million of boron which is cited in the bases to the specification. We find this change to be consistent with the approved reload licensing procedures of Reference 8 and therefore acceptable.

7.0 Summary

We conclude that the fuel system design, nuclear design, thermal-hydraulic design, transient accident analyses, and associated proposed TS changes for Peach Bottom 3 Cycle 6 operation are acceptable.

8.0 Environmental Consideration

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 impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

9.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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, and (3) 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: May 4, 1983

The following NRC personnel have contributed to this Safety Evaluation: M. Tokar, W. Brooks and S. Sun.

10.0 References

1. E. J. Bradley (PECo), letter to H. R. Denton (NRC), December 30, 1982.

2. R. A. Browning, "Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit 3, Reload 5," General Electric Report Y1003J01A54, December 1982.

3. A. N. Tschaeche, "Safety Evaluation of the General Electric Hybrid I Control Rod Assembly," General Electric Company Proprietary Report NEDE-22290, December 1982.

4. "Erratta and Addenda Sheet No. 5" to NEDO-24082, October 1982.

5. "Loss-of-Coolant Accident Analyses for Peach Bottom Atomic Power Station Unit 3," General Electric Company Report NEDO-24082, December 1977.

6. R. A. Blough (PECO), letter to J. F. Stolz (NRC), Subject: Peach Bottom Unit 3 Lead Test Assemblies, December 20, 1982.

7. NEDO-21363-4, Supplement 4, January 1977.

8. "General Electric Standard Application for Reactor Fuel," GESTAR-II, General Electric Report NEDE-24011 (latest approved version).

9. R. E. Engel (GE), letter to T. A. Ippolito (NRC), May 6, 1981.

10. R. E. Engel (GE), letter to T. A. Ippolito (NRC), May 28, 1981.

11. L. S. Rubenstein (NRC), memorandum for T. M. Novak, "Extension of General Electric Emergency Core Cooling Systems Performance Limits," June 25, 1981.

12. S. L. Daltroff (PECo), letter to J. F. Stolz (NRC), July 15, 1981.

13. P. Aldred, "BWR Control Rod Cobolt Alloy Replacement," Executive Summary, EPRI Report NP-2329-SY, March 1982.

14. A. N. Tschaeche (GE), letter to M. Tokar (NRC) with GE proprietary document titled "Hybrid Control Rod Licensing Presentation to the Nuclear Regulatory Commission, March 23, 1983," March 24, 1983.

15. "Proposed Peach Bottom Atomic Power Station Unit 2 Alternate Absorber Control Blade Test Program," General Electric Company Report NEDO-24231, Rev. 1, January 1980.

16. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154P, October 1978.

17. J. W. Gallagher (PECo), letter to J. R. Stolz (NRC), April 6, 1983.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 92 to Facility Operating License No. DPR-56, issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications (TSs) for operation of the Peach Bottom Atomic Power Station, Unit No. 3 (the facility) located in York County, Pennsylvania. The amendment is effective as of the date of issuance.

The amendment changes the TSs to permit Cycle 6 operation of the facility.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. 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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendment dated December 30, 1982, as supplemented April 6, 1983, (2) Amendment No. 92 to License No. DPR-56 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, NW., Washington, D.C., and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 4th day of May 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing

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

May 4, 1983

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Docket No. 50-278

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: **PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3**

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Amendment No. 92.
  Referenced documents have been provided PDR.

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

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