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Dockets Nos. 50-277
and 50-278

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. DPR-44 and Amendment No. 23 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Units Nos. 2 and 3. These amendments consist of changes to the Technical Specifications and are in response to your request dated April 14, 1976.

The amendment will correct certain inconsistencies and improve clarity in various sections of the Technical Specifications. In addition, revised wording of Technical Specification 3.6.A.1 on reactor vessel heatup and cooldown rates as requested by the Commission in our letter dated April 17, 1975 is included.

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

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 24 to DPR-44
2. Amendment No. 23 to DPR-56
3. Safety Evaluation
4. Federal Register Notice

cc w/encls:
See next page

SEE PREVIOUS YELLOW FOR CONCURRENCE

OFFICE	ORB#3	OELD	ORB#3			
SURNAME	TVerdery		GLear			
DATE	6/ /76	6/ /76	6/ /76			

Dockets Nos. 50-277
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2301 Market Street
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The amendment will correct certain inconsistencies and improve clarity in various sections of the Technical Specifications. In addition, revised wording of Technical Specification 3.6.A.1 on reactor vessel heatup and cooldown rates as requested by the Commission in our letter dated April 17, 1976 is included.

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

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 24 to DPR-44
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3. Safety Evaluation
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cc w/encls:
See next page

OFFICE➤	~~ORB#3~~	~~ORB#3~~	~~OELD~~	~~ORB#3~~		
SURNAME➤	CParrish	~~TVerdery~~		~~GLear~~		
DATE➤	6/ /76	6/ /76	6/ /76	6/ /76		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166 cc w/enclosures:

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Philadelphia, Pennsylvania 19101

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Albert R. Steel, Chairman
Board of Supervisors
Peach Bottom Township
R. D. #1
Delta, Pennsylvania 17314

Philadelphia Electric Company
ATTN: Mr. W. T. Ullrich
Peach Bottom Atomic
Power Station
Delta, Pennsylvania 17314

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. 24
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, (the licensees) dated April 14, 1976, 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; and

   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.

   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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: June 30, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 24

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace pages 111, 143, 170, 173, 174, and 195 with the attached revised pages.

3.3 and 4.3 BASES (Cont'd.)

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCHFR from becoming less than 1.0. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure 3.6.14) with the average response of all the drives as given in the above Specification, provide the required protection, and MCHFR remains greater than 1.0.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Peach Bottom.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds one control rod of a (5 x 5) twenty-five control rod array.

In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero and approximately 200 milliseconds later, control rod motion begins. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C. In addition the control rod drop accident has been analyzed in NEDO 10527 and its supplements 1 & 2 for the scram times given in Specification 3.3.C.

Surveillance requirement 4.3.C was originally written and used as a diagnostic surveillance technique during pre-operational and startup testing of Dresden 2 & 3 for the early discovery and identification of significant changes in drive scram performance following major changes in plant operation. The reason for the application of this surveillance was the unpredictable and degraded scram performance of drives at Dresden 2. The cause of the slower performances has been conclusively

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F increase (or decrease) in any one-hour period.

2. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Figure 3.6.1. The reactor vessel shall not be pressurized above 250 psig unless the reactor vessel temperatures are equal to or greater than 185°F if fuel is in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following temperatures shall be permanently logged at least every 15 minutes until the difference between any two readings taken over a 45 minute period is less than 5°F.

(a) Bottom head drain.

(b) Recirculation loops A and B.

2. Reactor vessel temperature and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented.

Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest

LIMITING CONDITIONS FOR OPERATION

3.7.A Primary Containment

3. Pressure Supression Chamber - Reactor Building Vacuum Breakers

a. Expect as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be 0.5 $\pm$ 0.25 psid.

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and .c, below.

b. One drywell-suppression chamber vacuum breaker may be non-fully closed so long as it is deter-

SURVEILLANCE REQUIREMENTS

4.7.A Primary Containment

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation including set point shall be checked for proper operation every refueling outage.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle once a month.

b. When it is determined that a vacuum breaker valve is inoperable for opening at a time when

3.7.A.6.c (cont'd.)

the affected reactor must be taken out of power operation. See the note at the end of this specification.

d. A 30 psig limit is the maximum containment repressurization allowable using the CAD system. Venting via the SBGT system to this stack must be initiated at 30 psig following the initial peak pressure of 49.1 psig.

NOTE: Prior to the end of the first refueling outage of each Unit, the normal containment inerting makeup system will serve as the post-LOCA Containment Atmosphere Dilution System. During that period, specifications 3.7.A.6.a. and 3.7.A.6.b. apply to the normal containment inerting makeup system. After the first refueling outage of each unit, specifications 3.7.A.6.a, 3.7.A.6.b. and 3.7.A.6.c. will apply to the system specifically designed for post-LOCA containment dilution, i.e., the CAD system.

4.7.A.6. (cont'd.)

**The CAD system H_2 and O_2 analyzers shall be tested for operability using standard bottled H_2 and O_2 once per month and shall be calibrated once per [illegible] months. The atmosphere analyzing system shall be functionally tested once per operating cycle in conjunction with specification 4.7.A.6.a. Should one of the two H_2 or O_2 analyzers serving the drywell or suppression pool be found inoperable the remaining analyzer of the same type serving the [illegible] compartment shall be tested for operability once per week until the defective analyzer is made operable. [illegible] the note at the end of specification [illegible].

NOTE: Prior to the end of the first refueling outage of each unit, the normal containment inerting makeup system will serve as the post-LOCA Containment Atmosphere Dilution System. During that period, specifications 4.7.A.6.a. and 4.7.A.6.b.,

LIMITING CONDITIONS FOR OPERATION

3.7.A.7. If the specifications of 3.7.A.1 through 3.7.A.5 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.A.6 (Cont'd.)

NOTE: apply to the normal containment inerting makeup system and only the portion of specification 4.7.A.6.c. marked with an * will apply. After the end of the first refueling outage for each unit, specifications 4.7.A.... 4.7.A.6.b., and 4.7.A.6.d. will apply to the CAD system and only that portion of specification 4.7.A.6.c. marked ** will apply.

Amendment No. 24

3.7.A & 4.7.A <u>BASES</u> (Cont'd.)

Due to the nitrogen addition, the pressure in the containment after a LOCA will increase with time. Under the worst expected conditions the containment pressure will reach 30 psig in approximately 18 days. If and when that pressure is reached, venting from the containment shall be manually initiated. The venting path will be through the Standby Gas Treatment system in order to minimize the off site dose.

Following a LOCA, periodic operation of the drywell and torus sprays will be used to assist the natural convection and diffusion mixing of hydrogen and oxygen.

UNITED STATES
NU EAR 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. 23
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, (the licensees) dated April 14, 1976, 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; and

   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.

   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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: June 30, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 23

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace pages 111, 143, 170, 173, 174 and 195 with the attached revised pages.

3.3 and 4.3 BASES (Cont'd.)

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCHFR from becoming less than 1.0. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure 3.6.14) with the average response of all the drives as given in the above Specification, provide the required protection, and MCHFR remains greater than 1.0.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Peach Bottom.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds one control rod of a (5 x 5) twenty-five control rod array.

In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero and approximately 200 milliseconds later, control rod motion begins. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C. In addition the control rod drop accident has been analyzed in NEDO 10527 and its supplements 1 & 2 for the scram times given in Specification 3.3.C.

Surveillance requirement 4.3.C was originally written and used as a diagnostic surveillance technique during pre-operational and startup testing of Dresden 2 & 3 for the early discovery and identification of significant changes in drive scram performance following major changes in plant operation. The reason for the application of this surveillance was the unpredictable and degraded scram performance of drives at Dresden 2. The cause of the slower performances has been conclusively

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F increase (or decrease) in any one-hour period.

2. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Figure 3.6.1. The reactor vessel shall not be pressurized above 250 psig unless the reactor vessel temperatures are equal to or greater than 185°F if fuel is in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following temperatures shall be permanently logged at least every 15 minutes until the difference between any two readings taken over a 45 minute period is less than 5°F.

(a) Bottom head drain.

(b) Recirculation loops A and B.

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Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest

LIMITING CONDITIONS FOR OPERA ON

3.7.A Primary Containment

3. Pressure Supression Chamber - Reactor Building Vacuum Breakers

a. Expect as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be 0.5 $\pm$ 0.25 psid.

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

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a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and .c, below.

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SURVEILLANCE REQUI ENTS

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4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle once a month.

b. When it is determined that a vacuum breaker valve is inoperable for opening at a time when

3.7.A.6.c (cont'd.)

the affected reactor must be taken out of power operation. See the note at the end of this specification.

d. A 30 psig limit is the maximum containment re-pressurization allowable using the CAD system. Venting via the SBGT system to this stack must be initiated at 30 psig following the initial peak pressure of 49.1 psig.

NOTE: Prior to the end of the first refueling outage of each Unit, the normal containment inerting makeup system will serve as the post-LOCA Containment Atmosphere Dilution System. During that period, specifications 3.7.A.6.a. and 3.7.A.6.b. apply to the normal containment inerting makeup system. After the first refueling outage of each unit, specifications 3.7.A.6.a, 3.7.A.6.b. and 3.7.A.6.c. will apply to the system specifically designed for post-LOCA containment dilution, i.e., the CAD system.

4.7.A.6.c. (cont'd.)

**The CAD system H_2 and O_2 analyzers shall be tested for operability using standard bottled H_2 and O_2 once per month and shall be calibrated once per 6 months. The atmosphere analyzing system shall be functionally tested once per operating cycle in conjunction with specification 4.7.A.6.a. Should one of the two H_2 or O_2 analyzers serving the drywell or suppression pool be found inoperable the remaining analyzer of the same type serving the same compartment shall be tested for operability once per week until the defective analyzer is made operable. See the note at the end of specification 4.7.A.

NOTE: Prior to the end of the first refueling outage of each unit, the normal containment inerting makeup system will serve as the post-LOCA Containment Atmosphere Dilution System. During that period, specifications 4.7.A.6.a. and 4.7.A.6.b,

Amendment No. 23

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A.6 (Cont'd.) NOTE: apply to the normal containment inerting makeup system and only the portion of specification 4.7.A.6.c. marked with an * will apply. After the end of the first refueling outage for each unit, specifications 4.7.A.6.a., 4.7.A.6.b., and 4.7.A.6.d. will apply to the CAD system and only that portion of specification 4.7.A.6.c. marked ** will apply.
3.7.A.7. If the specifications of 3.7.A.1 through 3.7.A.5 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.	

3.7.A & 4.7.A BASES (Cont'd.)

Due to the nitrogen addition, the pressure in the containment after a LOCA will increase with time. Under the worst expected conditions the containment pressure will reach 30 psig in approximately 18 days. If and when that pressure is reached, venting from the containment shall be manually initiated. The venting path will be through the Standby Gas Treatment system in order to minimize the off site dose.

Following a LOCA, periodic operation of the drywell and torus sprays will be used to assist the natural convection and diffusion mixing of hydrogen and oxygen.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 24 TO FACILITY LICENSE NO. DPR-44 AND

AMENDMENT NO. 23 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 UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

Introduction

By letter dated April 14, 1976, Philadelphia Electric Company (PECO) requested an amendment to Facility Operating Licenses Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station Units Nos. 2 and 3. The request involves the correction of certain inconsistencies and the improvement of clarity in various section of the Technical Specifications. Additionally, PECO has included the revised wording of Technical Specification 3.6.A.1 on reactor vessel heatup and cooldown rates as requested by the Commission in our letter dated April 17, 1975.

Evaluation

The changes proposed by the licensee involve purely editorial corrections and will not affect the operation of the facilities.

The incorporation of standardized wording on reactor vessel heatup and cooldown rates will remove an existing ambiguity such that the period over which the change in temperature is to be averaged can only be interpreted as one hour. Technical Specification 3.6.A.1 will now allow an instantaneous temperature change up to 100°F on the condition that no further changes take place during the succeeding one hour. This was the original intent of the Technical Specifications. We conclude that this change will effectively remove any ambiguity by specifying an averaging period of one hour. Since the change involves only nomenclature, there is no effect on normal plant operation or the consequences of any accident.

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change 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: June 30, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

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

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 24 and 23 to Facility Operating Licenses Nos. DPR-44 and DPR-56, respectively, 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 for operation of the Peach Bottom Atomic Power Station, Units Nos. 2 and 3, located in Peach Bottom, York County, Pennsylvania. The amendments are effective as of their date of issuance.

The amendment will correct certain inconsistencies and improve clarity in various sections of the Technical Specifications. In addition, vessel heatup and cooldown rates as requested by the Commission in our letter dated April 17, 1975, are included.

The application for the amendments 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 amendments.
Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendments dated April 14, 1976, (2) Amendment No. 24 to License No. DPR-44, (3) Amendment No. 23 to License No. DPR-56, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street N.W., Washington, D. C. and at the Martin Memorial Library, 159 E. Market Street, York, Pennsylvania 17401.

A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 30 day of June 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors