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Docket No. 50-278

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

Gentlemen:

In response to your request dated March 23, 1976, the Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station Unit No. 3.

The amendment consists of changes in the Technical Specifications to correct an error in the operating limit Minimum Critical Power Ratio (MCPR).

The Commission is deferring issuance of a similar amendment to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station Unit No. 2 pending our review of your application for license amendment dated March 22, 1976 which would authorize operation of Unit No. 2 following the first core refueling.

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. 18 to License DPR-56
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

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

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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 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 18
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 March 23, 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. After weighing the environmental aspects involved, 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 8, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 18

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace pages 17, 18, 54, 133a, 133b, 140a, 140b, 141a and 141b with the attached revised pages.

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Peach Bottom Atomic Power Station Units have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 3440 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7.1 of the FSAR. In addition, 3293 MWt is the licensed maximum power level of each Peach Bottom Atomic Power Station Unit, and this represents the maximum steady state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in NEDO 10802.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. Active coolant flow is equal to 88% of total core flow. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady state condition.

For analyses of the thermal consequences of the transients a MCPR of 1.26 is conservatively assumed to exist prior to initiation of the transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculating pumps.

In summary:

i. The abnormal operational transients were analyzed to a power level of 3440 MWt.

ii. The licensed maximum power level is 3293 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual trip settings are discussed in the following paragraphs.

A. Neutron Flux Scram

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (3293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to assure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum total peaking factor is greater than 2.63.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR $>$ 1.06 when the transient is initiated from MCPR $>$ 1.26.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; e.e., in the period of a month a maximum drift of 0.4% could occur, thus providing for adequate margin.

For the APRM system, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.1 and 4.1.2 indicates that two instrument channels have not been included in the latter table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable.

B. The peak heat flux is checked once per day to determine if the APRM scram requires adjustment. This will normally be done by checking the LPRM readings. Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check of the peak heat flux is adequate.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating twice a week using heat balance data and by calibrating individual LPRM's every 6 weeks, using TIP traverse data.

It is highly improbable that in actual operation with MTPF at 2.63 that MCPR will be as low as 1.26. Usually with peaking factors of this magnitude the peak occurs low in the core in a low quality region where the initial heat flux is very high. The MCPR design power shape (TPF = 2.43) assumes that the peak occurs higher in the core and represents the worst combination of individual peaking factor magnitude and shape, from a MCPR consideration that can be expected to occur in the core. Therefore, with TPF < 2.43 there are not technical specification requirements for calculating MCPR With TPF greater than 2.43 the daily requirement for calculating MCPR is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for caluclating MCPR when a limiting control pattern is approached insures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

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 Figure 3.5.1-C or 3.5.1-D, as applicable. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within prescribed limits within two (2) hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

3.5.J. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR \leq LHGR_d \ [1 - (\Delta P/P)_{max}(L/LT)]$$

$LHGR_d$ = Design LHGR = 18.5 kW/ft

$(\Delta P/P)_{max}$ = Maximum power spiking penalty = 0.026

LT = Total core length = 12.167 ft Unit 3

L = Axial position above bottom of core

4.5.I. Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during power 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.

LIMITING CONDITION OF OPERATION | SURVEILLANCE REQUIREMENT

3.5.J. Local LHGR (Cont'd.)

If at any time during power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

3.5.K. Minimum Critical Power Ratio (MCPR)

During power operation, MCPR shall be $\geq$1.26 at rated power and flow. Forc core flows other than rated the MCPR shall be 1.26 times k_f, where k_f is shown in Figure 3.5.1-E. If at any time during power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within prescribed limits within two (2) hours, the reactor shall be brough to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

4.5.K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.

J. Local LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.5.K are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06, and an analysis of abnormal operational transients presented in Reference 1. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The limiting transient which determines the required steady state MCPR limit is the rod withdrawal error transient. This transient yields the largest Δ MCPR. When added to the safety limit MCPR of 1.06 the required minimum operating limit MCPR of specification 3.5.K are obtained.

Two codes are used to analyze the rod withdrawal error transient. The first code simulates the three dimensional BWR core nuclear and thermal-hydraulic characteristics. Using this code a limiting control rod pattern is determined; the following assumptions are included in this determination:

(1) The core is operating at full power in the xenon-free condition.

(2) The highest worth control rod is assumed to be fully inserted.

(3) The analysis is performed for the most reactive point in the cycle.

(4) The control rods are assumed to be the worst possible pattern without exceeding thermal limits.

(5) A bundle in the vicinity of the highest worth control rod is assumed to be operating at the maximum allowable linear heat generation rate.

(6) A bundle in the vicinity of the highest worth control rod is assumed to be operating the minimum allowable critical power ratio.

The three-dimensional BWR code then simulates the core response to the control rod withdrawal error. The second code calculates the Rod Block Monitor response to the rod withdrawal error. This code simulates the Rod Block Monitor under selected failure conditions (LPRM) for the core response (calculated by the 3-dimensional BWR simulation code) for the control rod withdrawal.

The analysis of the rod withdrawal error for Peach Bottom Unit 3 considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor which is operating with the limiting control rod pattern as discussed above. This rod pattern is shown in Figure 7-6 of NEDO-21140.[1/]

A brief summary of the analytical method used to determine the nuclear characteristics is given in Section 5.3 of NEDO-20360.[2/]

L. References

1. "Peach Bottom Atomic Power Station Unit 3 Channel Inspection and Safety Analyses with Bypass Holes Plugged," NEDO-21140, Revision 1, March 1976.

2. General Electric BWR Generic Reload Application for 8x8 fuel, NEDO-20360, Revision 1, November 1974.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), August 1974.

4.5.K Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

4.5.L MCPR Limits for Core Flows Other than Rated

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of 1.26 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves shown in Figure 3.5.1-E were developed generically and are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow rate (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.5.1-E, are acceptable for Peach Bottom Unit 3 operation because the operating limit MCPR is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

4.5.M References

1. "Peach Bottom Atomic Power Station Unit 3 Channel Inspection and Safety Analyses with Bypass Holes Plugged," NEDO-21140, Revision 1.

2. General Electric BWR Generic Reload Application for 8 x 8 fuel, NEDO-20360, Revision 1, November 1974.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), August 1974.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 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 2

DOCKET NO. 50-277

Introduction

By letter dated March 23, 1976 Philadelphia Electric Company (PECO) requested a license amendment that would correct an error in the Technical Specifications for Peach Bottom Atomic Power Station Units Nos. 2 and 3. The error exists in the operating limit Minimum Critical Power Ratio (MCPR).

Discussion

In license amendments previously issued by the Commission to Peach Bottom Units Nos. 2 and 3 on November 28, 1975 (Amendment No. 15) and January 29, 1976 (Amendment No. 14) respectively, the Technical Specifications were modified to incorporate the General Electric Thermal Analysis Basis (GETAB). Subsequent to the issuance of these amendments the licensee informed us that, based upon refinements made in the General Electric Company's calculational model used for the Peach Bottom Units Nos. 2 and 3 GETAB transient analysis, the value for the operating limit MCPR for Peach Bottom Unit No. 3 has changed from 1.25 to 1.26. The operating limit MCPR value for Peach Bottom Unit No. 2 has also changed from 1.21 to 1.26 based on the same calculational model refinement; however, the Unit No. 2 license is not being amended at this time because PECO has submitted a new GETAB analysis by letter dated March 22, 1976 which is based on the 8x8 reload fuel loaded into the Unit No. 2 core during the first refueling. Accordingly, the Peach Bottom Unit No. 2 Technical Specifications will be modified to reflect this new analysis prior to startup following the current refueling outage.

Evaluation

The Control Rod Withdrawal Error Analysis results which are described in NEDO-21140 1/ as revised in March 1976, have been modified such that the

1/ General Electric, "Peach Bottom Atomic Power Station Unit 3 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged," NEDO-21140, Revision 1, dated March 1976.

maximum transient ΔCPR for that transient is now 0.20 instead of 0.15. In the original version of NEDO-21140 [2] the limiting operations transient was "Turbine Trip without Bypass without Two Pump Trip" which yielded a ΔCPR of 0.19. The results of the turbine trip transient have remained unchanged. Consequently, the limiting transient for Peach Bottom Unit No. 3 is now the "Control Rod Withdrawal Error".

The NRC staff has reviewed PECO's proposed amendment and has determined that (1) the revised operating limit MCPR of 1.26 is more conservative than the existing limit of 1.25 and (2) the change in this limit is based upon a refinement to a previously approved calculational model[3]. Therefore, the NRC staff concludes that operation of Peach Bottom Atomic Power Station Unit 3 with an operating limit MCPR of 1.26 is acceptable.

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 8, 1976

2/ General Electric, "Peach Bottom Atomic Power Station Unit 3 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged, "NEDO-21140, December, 1975.

3/ "Review and Evaluation of GETAB (General Electric Thermal Analysis Basis) for BWRs, "Division of Technical Review, Directorate of Licensing, United States Atomic Energy Commission, September, 1974.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-278

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 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 for opeation of the Peach Bottom Atomic Power Station, Unit 3. The amendment is effective as of its date of issuance.

The amendment consists of changes in the Technical Specifications to correct an error in the operating limit Minimum Critical Power Ratio (MCPR).

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 this amendment 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 23, 1976, (2) Amendment No. 18 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 N. W., Washington, D. C. and at the Martin Memorial Library, 159 E. Market Street, York, Pennsylvania 17401.

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 Operating Reactors.

Dated at Bethesda, Maryland, this 8 day of June 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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