March 18, 1991

Docket Nos. 50-277
and 50-278

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Mr. George J. Beck
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Beck

SUBJECT: MINIMUM CRITICAL POWER RATIO SAFETY LIMITS, PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 (TAC NOS. 79325 AND 79326)

The Commission has issued the enclosed Amendment Nos. 157 and 159 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3. These amendments consist of changes to the Technical Specifications in response to your application dated December 17, 1990 as supplemented on January 22, 1991.

These amendments would revise the Technical Specifications to revise Minimum Critical Power Ratio Safety Limits since the cores will be reloaded with a new fuel type, GE 8X8 NB, for Cycle 9 operation. The proposed amendments also involve miscellaneous administrative changes.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

Gene Y. Suh, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 157 to DPR-44
2. Amendment No. 159 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	:PDI-2/LA	:PDI-2/PM	:OGC	:PDI-2/D	:
NAME	:MO'Brien	:GSuh:rb	: MYoung	:WButler	:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 18, 1991

Docket Nos. 50-277
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Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
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These amendments would revise the Technical Specifications to revise Minimum Critical Power Ratio Safety Limits since the cores will be reloaded with a new fuel type, GE 8X8 NB, for Cycle 9 operation. The proposed amendments also involve miscellaneous administrative changes.

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Gene Y. Suh

Gene Y. Suh, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 157 to DPR-44
2. Amendment No. 159 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page

cc:

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Power Plant and Environmental
Review Division
Department of Natural Resources
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Annapolis, Maryland 21401

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. 157
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, et. al. (the licensee) dated December 17, 1990 as supplemented on January 22, 1991, 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 or safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-44 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 157 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of startup in Cycle 9.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 18, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 157

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove	Insert
9	9
17	17
24	24
140a	140a
140b	140b
140c	140c
256a	256a

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objectives:

The objective of the Safety Limits is to establish limits which assure the integrity of the fuel cladding.

Specification:

A. Reactor Pressure $\geq$ 800 psia and Core Flow $\geq$ 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.06 for two recirculation loop operation, or 1.07 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objectives:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.58W + 62\% - 0.58\ \Delta W$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculating flow rate in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

2.1 BASES: 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 or above the thermal power condition required by Regulatory Guide 1.49. 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. Conservatism incorporated into the transient analyses is documented in References 2 and 3.

2.1 BASES: (Cont'd)

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO 10802, February 1973.

2. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

3. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

3.5. BASES (Cont'd)

J. Local LHGR

This specification assures that the linear heat generation rate in any 8X8 fuel rod is less than the design linear heat generation. The maximum LHGR 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 at the design LHGR below 25% rated thermal power, the peak local LHGR must be a factor of approximately ten (10) greater than the average LHGR 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 are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and References 7 and 10. 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 violated 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 transients evaluated are as described in References 7 and 10.

3.5.K. BASES (Cont'd)

The largest reduction in critical power ratio is then added to the fuel cladding integrity safety limit MCPR to establish the MCPR Operating Limit for each fuel type.

Analysis of the abnormal operational transients is presented in References 7 and 10. Input data and operating conditions used in this analysis are shown in References 7 and 10 and in the Supplemental Reload Licensing Analysis.

3.5.L. Average Planar LHGR (APLHGR), Local LHGR and Minimum Critical Power Ratio (MCPR)

In the event that the calculated value of APLHGR, LHGR or MCPR exceeds its limiting value, a determination is made to ascertain the cause and initiate corrective actions to restore the value to within prescribed limits. The status of all indicated limiting fuel bundles is reviewed as well as input data associated with the limiting values such as power distribution, instrumentation data (Traversing In-Core Probe - TIP, Local Power Range Monitor - LPRM, and reactor heat balance instrumentation), control rod configuration, etc., in order to determine whether the calculated values are valid.

In the event that the review indicates that the calculated value exceeding limits is valid, corrective action is immediately undertaken to restore the value to within prescribed limits. Following corrective action, which may involve alterations to the control rod configuration and consequently changes to the core power distribution, revised instrumentation data, including changes to the relative neutron flux distribution, for up to 43 in-core locations is obtained and the power distribution, APLHGR, LHGR and MCPR calculated. Corrective action is initiated within one hour of an indicated value exceeding limits and verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication.

In the event that the calculated value of APLHGR, LHGR or MCPR exceeding its limiting value is not valid, i.e., due to an erroneous instrumentation indication, etc., corrective action is initiated within one hour of an indicated value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

3.5.L. BASES (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limits value predominantely occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss-of-Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate.

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7 and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

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

5. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter, G. L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

6. DELETED.

7. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

8. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, NEDO-24081, December 1977, and for Unit 3, NEDO-24082, December 1977.

9. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, Supplement 1, NEDE-24081-P, November 1986.

10. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

6.9.1 Routine Reports (Cont'd)

(2) PECo-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECo)"

(3) PECo-FMS-0003-A, "Steady-State Fuel Performance Methods Report"

(4) PECo-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis"

(5) PECo-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"

(6) PECo-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations"

(3) The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(4) The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be submitted upon issuance for each Operating Cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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. 159
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 17, 1990 as supplemented on January 22, 1991, 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 or 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. 159, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of startup in Cycle 9.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 18, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 159

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 areas are indicated by marginal lines.

Remove	Insert
9	9
17	17
24	24
140a	140a
140b	140b
140c	140c
256a	256a

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objectives:

The objective of the Safety Limits is to establish limits which assure the integrity of the fuel cladding.

Specification:

A. Reactor Pressure $\geq$ 800 psia and Core Flow $\geq$ 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.06 for two recirculation loop operation, or 1.07 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objectives:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

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$S \leq 0.58W + 62\% - 0.58\ \Delta W$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculating flow rate in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

2.1 BASES: 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 or above the thermal power condition required by Regulatory Guide 1.49. 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. Conservatism incorporated into the transient analyses is documented in References 4 and 5.

2.1 BASES: (Cont'd)

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO 10802, February 1973.

2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", NEDO 24154 and NEDE 24154-P, Volumes I, II, and III.

3. "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors NEDO-24154 and NEDE 24154-P, Volumes I, II, and III.

4. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

5. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

3.5. BASES (Cont'd)

J. Local LHGR

This specification assures that the linear heat generation rate in any 8X8 fuel rod is less than the design linear heat generation. The maximum LHGR 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 at the design LHGR below 25% rated thermal power, the peak local LHGR must be a factor of approximately ten (10) greater than the average LHGR 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 are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and References 7 and 10. 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 violated 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 transients evaluated are as described in References 7 and 10.

3.5.K. BASES (Cont'd)

The largest reduction in critical power ratio is then added to the fuel cladding integrity safety limit MCPR to establish the MCPR Operating Limit for each fuel type.

Analysis of the abnormal operational transients is presented in References 7 and 10. Input data and operating conditions used in this analysis are shown in References 7 and 10 and in the Supplemental Reload Licensing Analysis.

3.5.L. Average Planar LHGR (APLHGR), Local LHGR and Minimum Critical Power Ratio (MCPR)

In the event that the calculated value of APLHGR, LHGR or MCPR exceeds its limiting value, a determination is made to ascertain the cause and initiate corrective action to restore the value to within prescribed limits. The status of all indicated limiting fuel bundles is reviewed as well as input data associated with the limiting values such as power distribution, instrumentation data (Traversing In-Core Probe - TIP, Local Power Range Monitor - LPRM, and reactor heat balance instrumentation), control rod configuration, etc., in order to determine whether the calculated values are valid.

In the event that the review indicates that the calculated value exceeding limits is valid, corrective action is immediately undertaken to restore the value to within prescribed limits. Following corrective action, which may involve alterations to the control rod configuration and consequently changes to the core power distribution, revised instrumentation data, including changes to the relative neutron flux distribution, for up to 43 in-core locations is obtained and the power distribution, APLHGR, LHGR and MCPR calculated. Corrective action is initiated within one hour of an indicated value exceeding limits and verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication.

In the event that the calculated value of APLHGR, LHGR or MCPR exceeding its limiting value is not valid, i.e., due to an erroneous instrumentation indication, etc., corrective action is initiated within one hour of an indication value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

3.5.L. <u>BASES</u> (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominantely occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss-of-Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate.

3.5.M. <u>References</u>

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7 and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

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

5. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter, G. L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

6. DELETED.

7. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

8. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, NEDO-24081, December 1977, and for Unit 3, NEDO-24082, December 1977.

9. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, Supplement 1, NEDE-24081-P, November 1986, and for Unit 3, NEDE-24082-P, December 1987.

10. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

6.9.1 Routine Reports (Cont'd)

(2) PECo-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECo)"

(3) PECo-FMS-0003-A, "Steady-State Fuel Performance Methods Report"

(4) PECo-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis"

(5) PECo-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"

(6) PECo-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations"

(3) The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(4) The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be submitted upon issuance for each Operating Cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 157 AND 159 TO FACILITY OPERATING

LICENSE NOS. DPR-44 and 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 NOS. 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By letter dated December 17, 1990, as supplemented on January 22, 1991, Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company and Atlantic City Electric Company, submitted a request for changes to the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, Technical Specifications (TS). The requested changes would revise the Technical Specifications (TS) to revise Minimum Critical Power Ratio Safety Limits since the cores will be reloaded with a new fuel type, GE8X8NB (or simply GE9B fuel), for Cycle 9 operation. The proposed amendments also involve miscellaneous administrative changes.

2.0 EVALUATION

The use of GE9B fuel in Units 2 and 3 requires MCPR Safety Limits not less than 1.06 for two-loop operation and 1.07 for single loop operation based on the approved GE methodologies. The Unit 2 Cycle 9 consists of 148 fuel assemblies of GE9B, 12 lead test assemblies (LTAs) and other irradiated fuel assemblies from previous cycles. The Unit 3 Cycle 9 is proposed to have 256 fuel assemblies of GE9B and other irradiated fuel assemblies from previous cycles. The 12 LTAs in Unit 2 Cycle 9 core are four assemblies of GE11, four assemblies of ABB ATOMS's SVEA-96, and four assemblies of ANF's 9x9-9X+.

MCPR Safety Limit

For two-loop operation, the licensee used the approved methodology "General Electric BWR Thermal Analysis Basis (GETAB): Date, Correlation and Design Application," NEDO-10958-A, to determine that the MCPR Safety Limit is 1.06 for Peach Bottom, Units 2 and 3. For single loop operation, the licensee determined that the MCPR Safety Limit is 1.07 based on the approved methodology. The MCPR Safety Limit for irradiated fuel from previous cycles continues to be 1.04. Since the licensee used the approved methodologies to determine the MCPR Safety Limits, we consider the licensee's analyses acceptable.

LTAs

The licensee stated that the LTAs will be located in non-limiting positions such that there is no impact on the core-wide MCPR Safety Limit. The LTAs are designed to be compatible to the reload core. We thus consider that these 12 LTSs are acceptable for Unit 2 Cycle 9 core.

Technical Specifications Changes

Section 1.1 Safety Limit, Units 2 and 3

The MCPR Safety Limits are modified to reflect the new safety limit, 1.06 for two-loop operation and 1.07 for single loop operation. Since the new MCPR safety limits were based on the approved methodologies, we consider these changes to be acceptable.

Section 2.1 and Section 3.5 Bases, Units 2 and 3

The References in the TS Bases are modified to reflect the use of approved methodologies. We consider these changes to be acceptable.

Miscellaneous Administrative Changes

As documented in a June 15, 1990 letter to the licensee, the staff completed its review of the licensee's topical report PECO-FMS-006, "Methods for Performing BWR Reload Safety Evaluations," and concluded the topical report, the methodology, and PECo's use of the methodology are acceptable. The proposed change to TS 6.9.1 reflects the staff's approval of PECo-FMS-0006 and the licensee's issuance of an approved version of the topical report, and thus is acceptable. The remaining proposed changes are editorial in nature or correct typographical errors and are acceptable.

We have reviewed the licensee's submittal of Technical Specification changes for Peach Bottom Units 2 and 3. Based on the acceptable analyses and use of approved methodologies, we conclude that these Technical Specification changes are acceptable for Peach Bottom Units 2 and 3.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATIONS

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The amendments also involve changes to recordkeeping and reporting requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: S. L. Wu

Date: March 18, 1991