3/22/78

Distribution
Docket
ORB #3
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NRC PDR
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BScharf (10)
JMcGough
DEisenhut
ACRS (16)
OPA (CMiles)
DRoss
TBAbernathy
JRBuchanan

Docket No. 50-277

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. 40 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. The amendment consists of changes to the Technical Specifications and is in response to your request dated December 19, 1977, as supplemented January 17 and February 17, 1978.

The amendment revises the Technical Specifications to reflect the reevaluation of the Emergency Core Cooling System (ECCS) cooling performance submitted in accordance with the March 11, 1977 Order for Modification of License and Exemption from the requirements of 10 CFR 50.46.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 40
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

OFFICE➤	ORB #3	ORB #3	~~OELD~~	ORB #3	
SURNAME➤	SSheppard	DVerrelli:mjf	Cutchin	GLear	
DATE➤	3/8/78	3/9/78	3/14/78	3/ /78	

NRC FORM 318 (9-76) NRCM 0240

cc:

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

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
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 December 19, 1977, as supplemented January 17 and February 17, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 40, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl W. Reuse /for

Victor Stello, Jr., Acting Assistant Director for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 22, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 40

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Revise Appendix A as follows:

Remove	Replace
iv	iv
133a	133a
140	140
140d	140d
140e	140e
142	142
142a	142a
142b	142b
142c	142c
142e	142e

Marginal lines indicate revised area

LIST OF FIGURES

LIMITING CONDITIONS FOR OPERATION

3.5.I Average Planar LHGR

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.5.1.A, B, C, D, & F, as applicable. If at any time during operation it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore APLHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is with the prescribed limits.

3.5.J Local LHGR

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

$$LHGR \leq LHGRd\ [1-(\Delta P/P)max\ (L/LT)]$$

LHGRd = Design LHGR
= 18.5 kW/ft for 7x7 fuel
13.4 kW/ft for 8x8 fuel

$(\Delta P/P)max$ = Maximum power spiking penalty
= 0.026 for 7x7 fuel
= 0.022 for 8x8 fuel

LT = Total core length = 12 ft.

L = Axial position above bottom of core

SURVEILLANCE REQUIREMENTS

4.5.I Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure shall be checked daily during reactor operation at ≥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 ≥25% rated thermal power.

3.5 BASES (Cont'd.)

H. Engineering Safeguards Compartments Cooling and Ventilation

One unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicate that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured. Ventilation associated with the High Pressure Service Water Pumps is also associated with the Emergency Service Water pumps, and is specified in Specification 3.9.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR Part 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent, secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The Technical Specification APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figure 3.5.1-A, B, C, D, and F.

The calculational procedure used to establish the APLHGR shown on Figures 3.5.1.A, B, C, D, and F is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. A complete discussion of each code employed in the analysis is presented in Reference 4. Input and model changes in the Peach Bottom loss-of-coolant analysis which are different from the previous analyses performed with Reference 4 are described in detail in Reference 8. These changes to the analysis include: (1) consideration of the counter current flow limiting (CCFL) effect, (2) corrected code inputs, and (3) the effect of drilling alternate flow paths in the bundle lower tie plate.

3.5.L BASES (Cont'd.)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominately 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.

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. "General Electric Boiling Water Reactor Reload-2 License Amendment for Peach Bottom Atomic Power Station Unit 2," NEDO-21578, February 1977.

7. General Electric BWR Generic Reload Application for 8x8 fuel, NEDO-20360, Revision 1, Supplement 4, April 1976.

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

TABLE 3.5-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

PLANT PARAMETERS:

Core Thermal Power	3440 MWt which corresponds to 105% of rated steam flow
Vessel Steam Output	14.05×10^6 lbm/h which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area For Large Breaks -	
Discharge	1.9 ft² (DBA)
Suction	4.1 ft²
Assumed Number of Drilled Bundles	360

FUEL PARAMETERS:

Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (KW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio
7x7, Type 2	7 x 7	18.5	1.5	1.2
7x7, Type 3	7 x 7	18.5	1.5	1.2
8x8, Type H	8 x 8	13.4	1.4	1.2
8x8, Type L	8 x 8	13.4	1.4	1.2
8x8 LTA	8 x 8	13.4	1.4	1.2

A more detailed list of input to each model and its source is presented in Section II of Reference 5.

PEACH BOTTOM UNIT 2

7x7 Fuel, Type 3

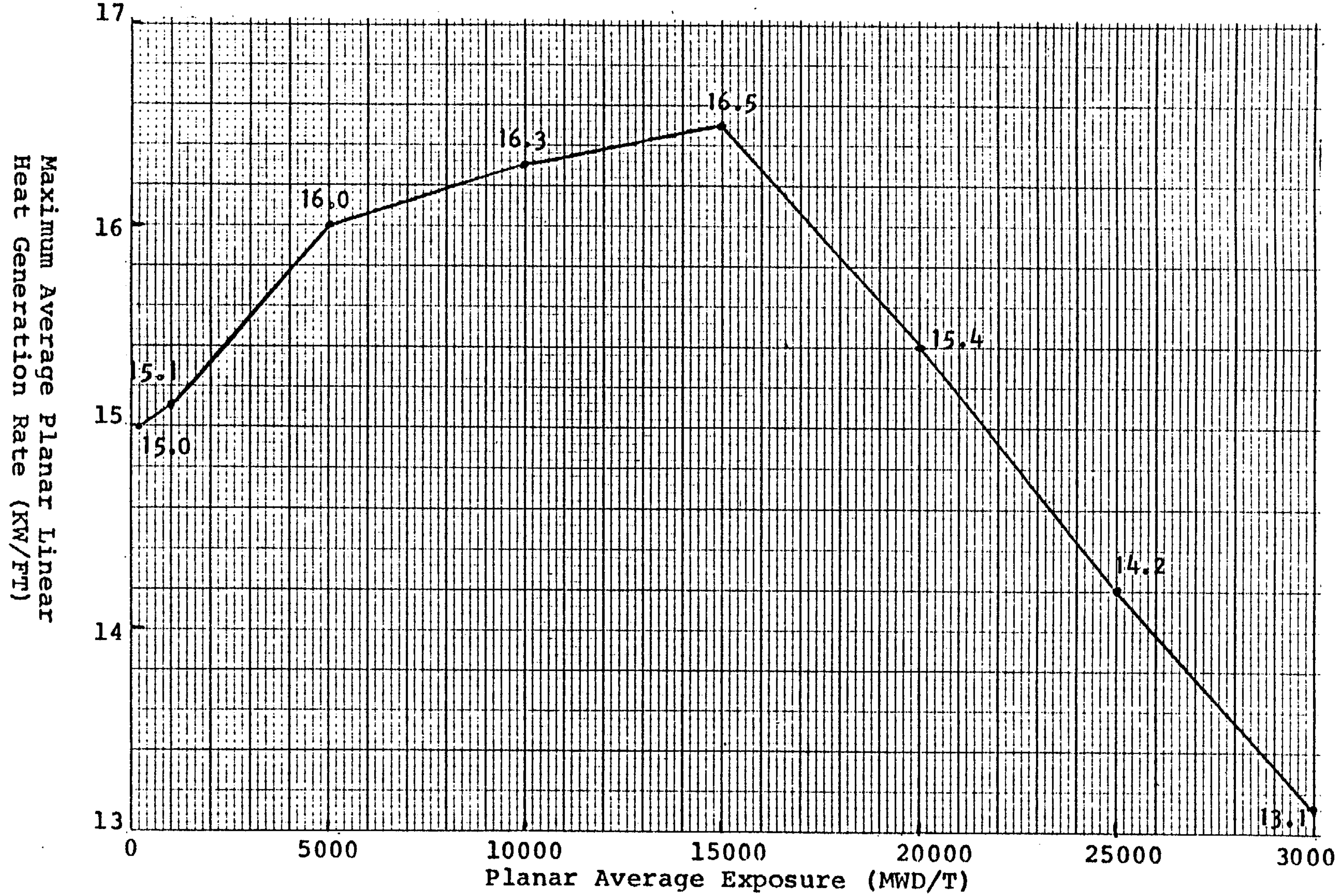

Figure 3.5.1.A Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

PEACH BOTTOM UNIT 2

7x7 Fuel, Type 2

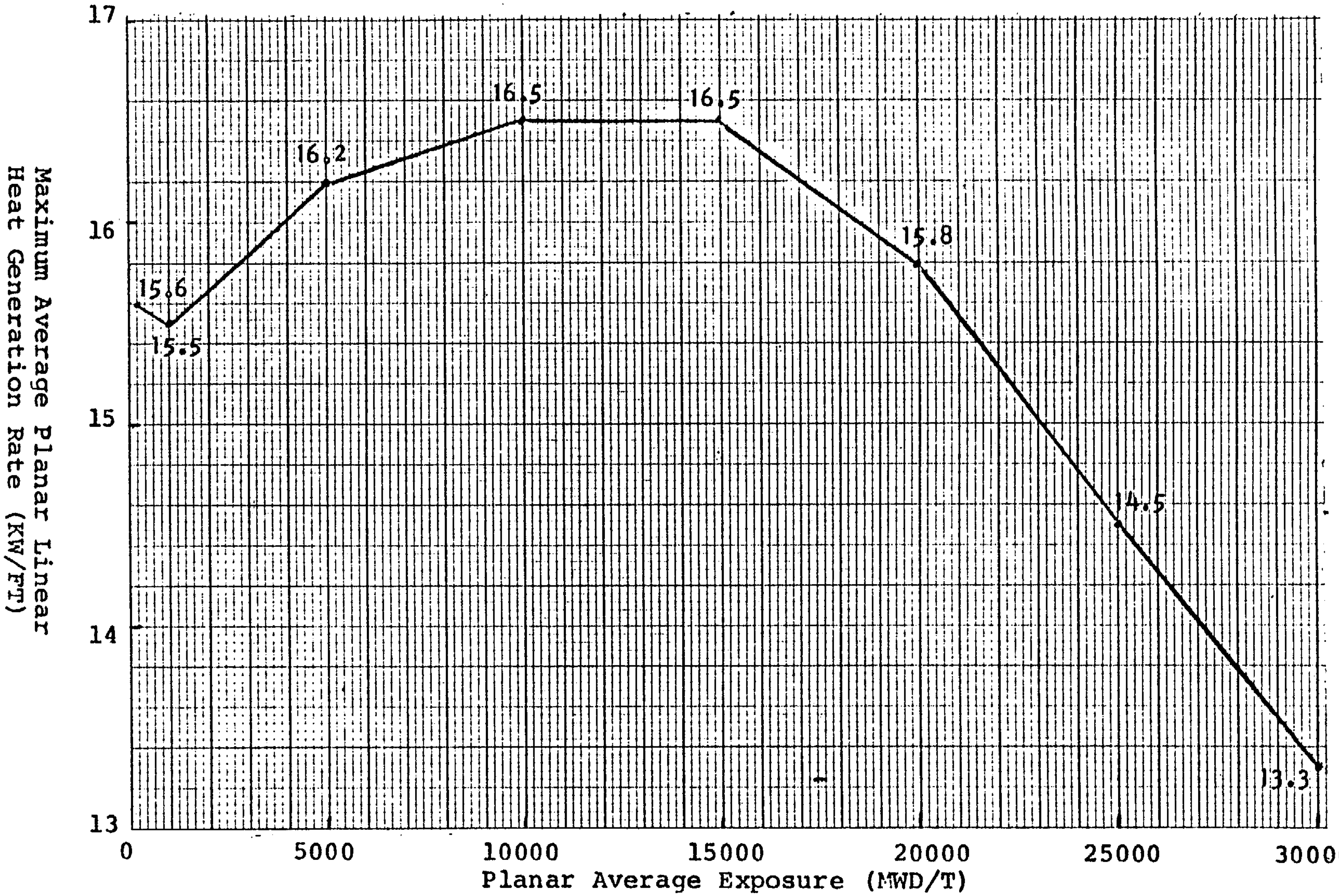

Figure 3.5.1.B Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

PEACH BOTTOM UNIT 2

8x8 Fuel, Type H - 80 mil & 100 mil

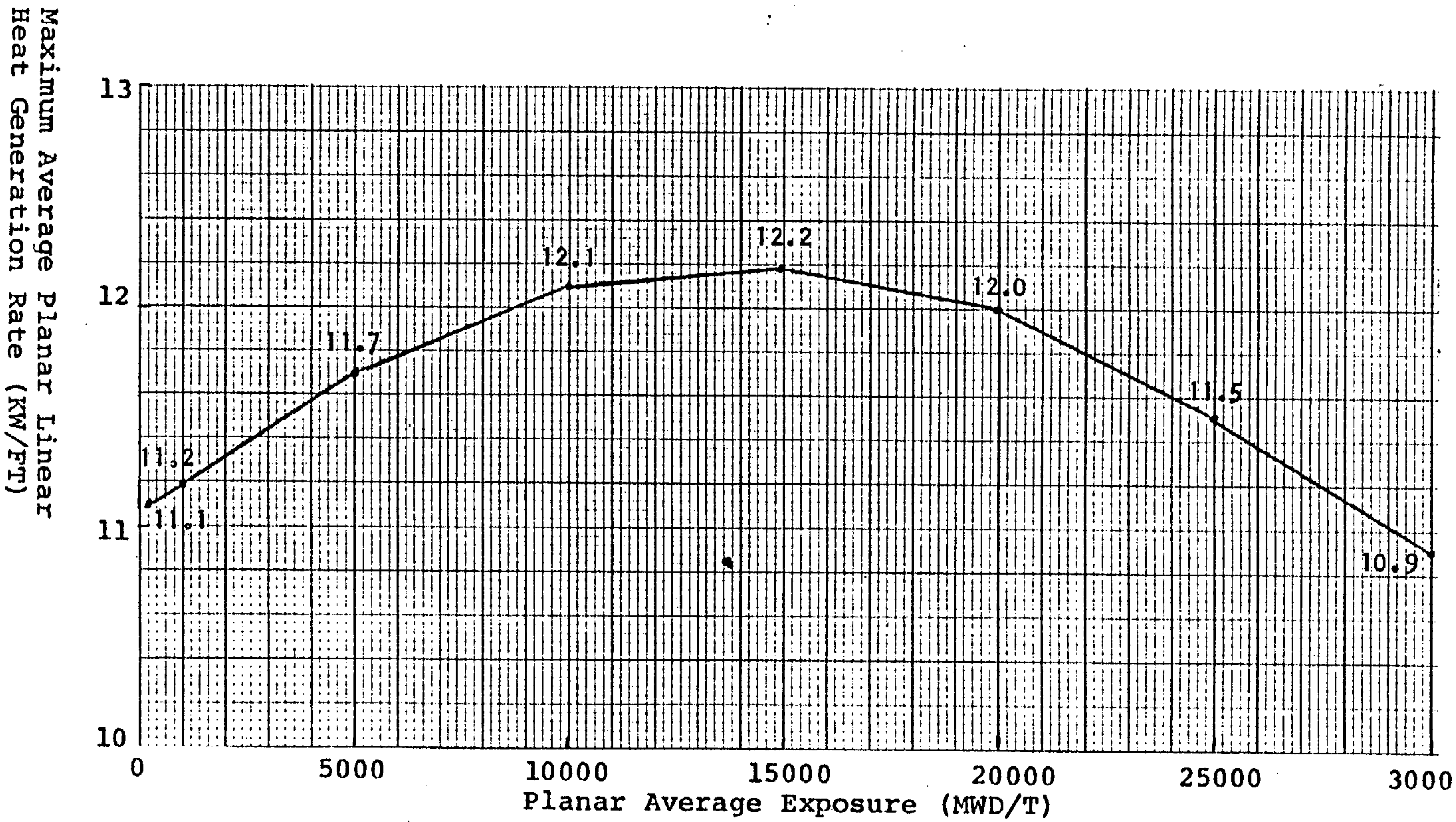

Figure 3.5.1.C   Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

PEACH BOTTOM UNIT 2

8x8 Fuel, Type L

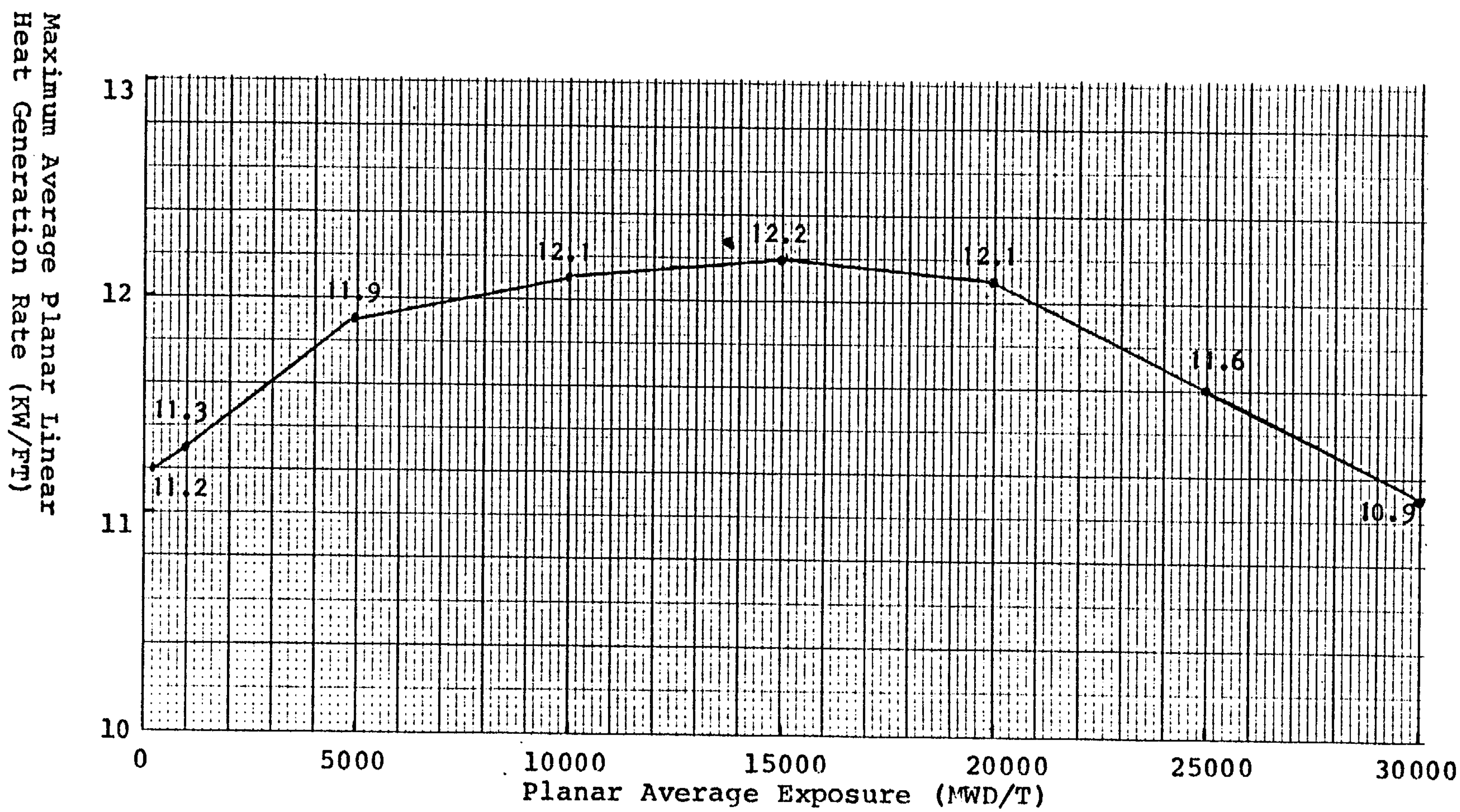

Figure 3.5.1.D Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

PEACH BOTTOM UNIT 2

8x8 LTA Fuel

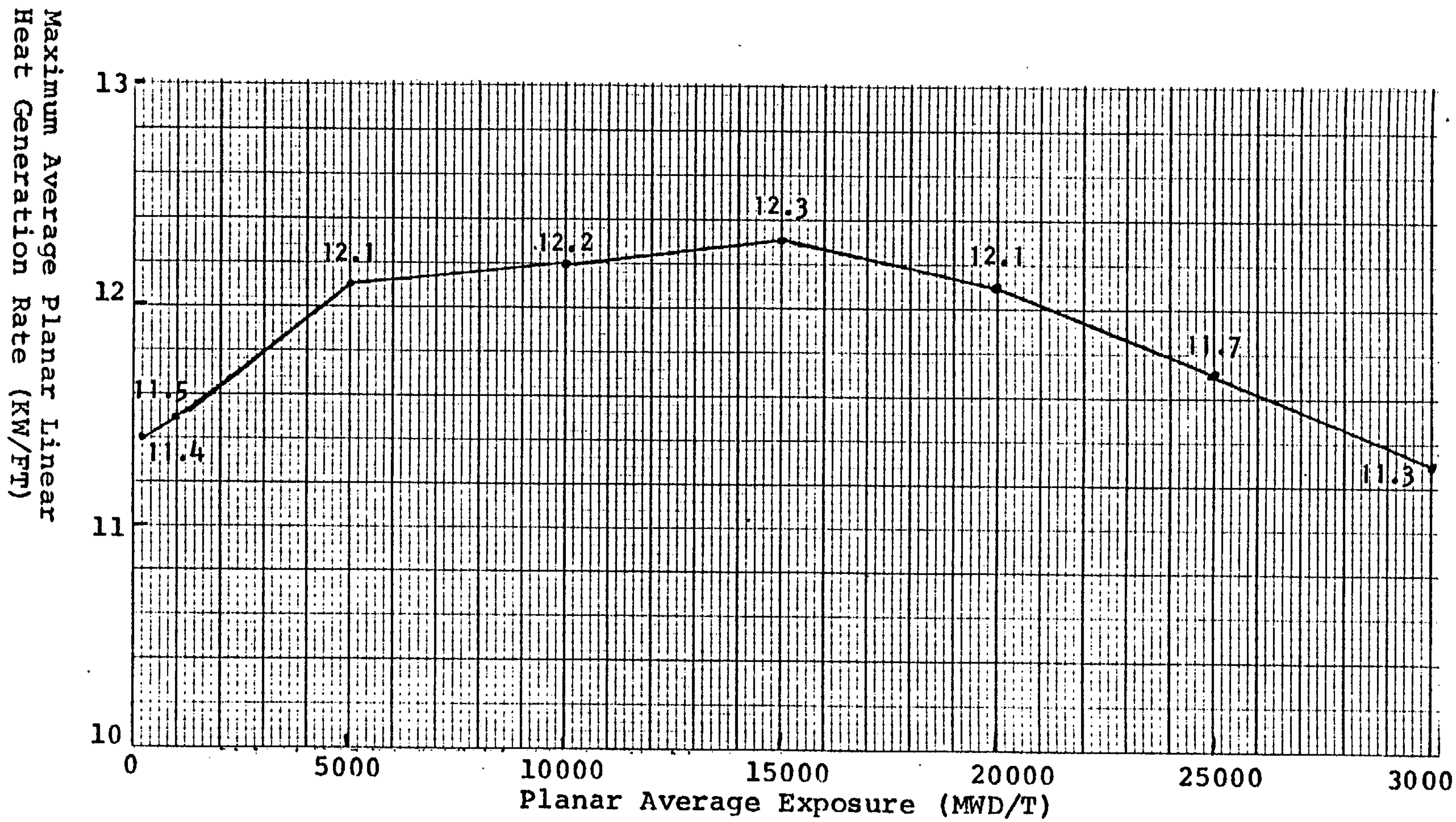

Figure 3.5.1.F Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 40 TO

FACILITY LICENSE NO. DPR-44

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNIT NO. 2

DOCKET NO. 50-277

INTRODUCTION

By letter dated December 19, 1977, as supplemented by letters dated January 17, 1978 and February 17, 1978, Philadelphia Electric Company (PECO) requested changes to the Technical Specifications in Appendix A to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station Unit No. 2 (PBAPS 2). These submittals satisfied the Commission's March 11, 1977 Order for Modification of License and Exemption to the requirements of 10 CFR 50.46. This licensing action was noticed in the FEDERAL REGISTER on February 2, 1978 (43 FR 4469).

BACKGROUND INFORMATION

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment "...the Licensee shall submit a reevaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results and assumptions. The licensee satisfied the requirements of this Order by previous submittals for reload licensing amendments.(1)(2)

In December of 1976 the NRC staff was informed that certain input errors and computer code errors had been made in the evaluations that were provided under the requirements described above. An Order was issued to PECO on March 11, 1977, requiring that corrected, revised calculations fully conforming to the requirements of 10 CFR 50.46 were to be provided for the PBAPS 2 as soon as possible. Such corrected analyses were provided for the present core in Reference 3. The corrected analyses included correction of all input errors previously made and correction of all computer code errors. The corrected analyses were performed using a calculational model which contains several model changes approved by the NRC staff in a Safety Evaluation issued April 12, 1977.

EVALUATION

We have reviewed the corrected analyses submitted in Reference 3, and the resulting Technical Specification changes submitted in Reference 4. We conclude that the PBAPS 2 will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: 1) it is operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Figures 3.5.1.A, 3.5.1.B, 3.5.1.C, 3.5.1.D, and 3.5.1.F of Reference 4; and 2) when it is operated at a Minimum Critical Power Ratio (MCPR) equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss-of-Coolant Accident).

The analyses submitted in Reference 3 provide all information requested in the NRC letter to GE on June 30, 1977 regarding number of breaks to be analyzed, documentation to be provided, etc. for the new analyses. These analyses for PBAPS 2 reference the lead plant (James A. Fitzpatrick Nuclear Power Plant) analyses for BWR/4 plants with the low-pressure-coolant-injection system modification.(5) The following description is provided of particular features of the analyses which are different from the lead plant, and the reason underlying those differences.

The break spectrum (i.e., PCT vs. break size) for the lead plant showed that the particular break producing the highest PCT for the lead plant was a recirculation pump discharge line break having an area approximately 80% as large as the largest discharge line break (5). However, the break spectrum for PBAPS 2 showed that the particular break producing the highest PCT is the largest (100%) discharge line break.

The SER for the lead plant (6) (which is incorporated by reference in this SER for PBAPS 2) explains the reasons why the discharge break location is limiting for that plant. As explained more fully in that SER, the largest break in the largest pipe would normally be expected to be limiting (the largest pipe is the suction pipe). However, the LPCI modification (also explained more fully in the lead plant SER) results in at least one loop of the LPCI system being available to help mitigate the consequences of suction pipe breaks even with the worst assumed single failure; but, due to certain piping and valve locations, with certain single failure assumptions, no LPCI system is available for the smaller, discharge line break. This results in a "tradeoff" of "compensating effects" situation where a larger, normally more severe break (suction line) has more ECCS available to mitigate its consequences, while a smaller, normally less severe break (discharge line) has less ECCS. The lead plant SER states that in most cases, this "tradeoff" results in the discharge location being limiting, as it is for Fitzpatrick and PBAPS 2.

In order to justify that the largest discharge line break is limiting for PBAPS 2 it is necessary to determine that no discharge or suction crack size that was not specifically analyzed for PBAPS 2 could be more limiting than the discrete sizes that were specifically analyzed.

The same arguments presented in the lead plant SER(6) regarding PCT vs. break size also apply to PBAPS 2. For PBAPS 2 the uncovery-time-interval vs. discharge break area curve peaks at 66% of the largest discharge line break's area. For suction breaks, the uncovery-time-interval vs. suction break area curve peaks at 100% of the largest suction line break's area.

Uncovery-time-interval is generally the single most important "time" in determining ultimate PCT. However, two other times that significantly affect PCT are departure-from-nucleate-boiling (DNB) time and uncovery time. Both of these times occur earlier as break size is increased; earlier DNB and earlier uncovery times each cause PCT to increase due to earlier loss of heat removal capability.

Therefore, for suction line breaks, all three "times" (uncovery-time-interval, DNB time, and uncovery time) are each individually at their value which would cause highest PCT at the same size (largest) break. Thus the largest suction line break would clearly have the highest PCT of any suction line break. This largest suction break's PCT was calculated to be 2148°F for PBAPS 2. For discharge line breaks, one of these "times" (uncovery-time-interval) would tend to cause the highest PCT at 66% of the largest discharge line's break area; the other two times (DNB and uncovery) would tend to cause highest PCT for 100% of the largest discharge line's break area. Specific calculations for these two breaks for PBAPS 2 have shown the "66%" break's PCT to be 2187°F, and the "100%" discharge line break's PCT to be 2197°F.

As illustrated in Figure 6a of Reference 3, the uncovery-time-period vs. discharge break area curve peaks very sharply at "66%"; and change to a slightly larger or smaller break area would cause a shift to a significantly shorter uncovery-time-period which would overcompensate for any possible effects on PCT in the other direction due to the size change (i.e., changes in DNB time or uncovery time). Between 80% and 100% the uncovery-time-period increases and the break at 100% results in the largest period for which the hot node remains uncovered. Over this range the 100% break results in the highest calculated PCT since, if two breaks have similar times for which the hot node remains uncovered, then the larger of the two breaks will be limiting since it would have an earlier uncovery and earlier DNB time (i.e., the larger break would have the more severe blowdown heat transfer analysis).

We therefore conclude, for the reasons stated above, that the most limiting break is the largest discharge line break for PBAPS 2. That break was used to generate the above referenced MAPLHGR limits, which we therefore find acceptable as stated previously.

Environmental Consideration

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

Conclusion

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

Dated: March 22, 1978

References

1. "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal with Partial Installation of the Alternate Flow Path for Peach Bottom Atomic Power Station Unit 2." License No. DPR-44, Docket No. 50-277. NEDO-21172, Revision 1. March 1976. Appendix A to NEDO-20360, Revision 1, Supplement 3, September, 1975.

2. "General Electric Boiling Water Reactor Reload No. 2 Licensing Amendment For Peach Bottom Atomic Power Station Unit 2." License No. DPR-44, Docket No. 50-277. NEDO-21578, February, 1977.

3. NEDO-24081, LOCA Analysis for Peach Bottom Atomic Power Station Unit 2, General Electric Co. December, 1977.

4. Letter to Director of Nuclear Reactor Regulation, USNRC, from E. J. Bradley, Philadelphia Electric Co., Application for Amendment of Facility Operating License DPR-44, December 19, 1977.

5. Letter to R. W. Reid, NRC, from George T. Berry, Power Authority of the State of New York, James A. Fitzpatrick Nuclear Power Plant ECCS Analysis, July 29, 1977 transmitting NEDO 21662-1 Loss of Coolant Accident Analysis Report for James A. Fitzpatrick Nuclear Power Plant, June 1977.

6. Letter to George T. Berry, Power Authority of the State of New York, from R. W. Reid, NRC, September 16, 1977 transmitting SER in support of Amendment 30 to DPR-59 for the James A. Fitzpatrick Nuclear Power Plant.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-277

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

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 40 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 operation of the Peach Bottom Atomic Power Station Unit No. 2. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to reflect the reevaluation of the Emergency Core Cooling System (ECCS) cooling performance submitted in accordance with the March 11, 1977 Order for Modification of License and Exemption from the requirements of 10 CFR 50.46.

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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on February 2, 1978 (43 FR 4469). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action. The Commission has determined that the issuance of this amendment will not result in any significant impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact 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 December 19, 1977 as supplemented January 17 and February 17, 1978, (2) Amendment No. 40 to License No. DPR-44, 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 Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. 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 22 day of March 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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