JUNE 1 4 1978

Docket No. 50-278

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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. 42 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Unit No. 3. The amendment consists of changes to the Technical Specifications and is in response to your request dated May 16, 1978.

The amendment modifies the Technical Specifications to incorporate exposure-dependent Minimum Critical Power Ratio (MCPR) operating limits for cycle 3 operation.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 42 to DPR-56
2. Safety Evaluation
3. Notice

cc w/enclosure:
See next page

OFFICE➤	ORB#3	ORB#3	~~OELD~~	ORB#3		RSB
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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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. 42
License No. DPR-56

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

   A. The application for the amendment by Philadelphia Electric Company, et al, (the licensee), dated May 16, 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-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. 42, 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

George Lear

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 14, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 42

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

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

Remove	Replace
vi	vi
18	18
33	33
133b	133b
-	133c
140a	140a
140b	140b
140c	140c
140d	140d
157	157

LIST OF TABLES

2.1 BASES (Cont'd.)

For analyses of the thermal consequences of the transients a MCPR equal to or greater than the operating limit MCPR given in Specification 3.5.K is conservatively assumed to exist prior to initiation of the limiting 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. 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, except for the End-of-Cycle Load Rejection transient which was analyzed at a power level of 3293 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.

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The pressure relief system for each unit at the Peach Bottom Atomic Power Station has been sized to meet two design bases. First, the total capacity of the safety/relief valves and safety valves has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4.1 of subsection 4.4 of the FSAR which states that the nuclear system safety/relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME Code requirements are presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report submitted in Appendix K.

Eleven safety/relief valves and two safety valves have been installed on Peach Bottom Unit 3. The analysis of the worst overpressure transient, (3-second closure of all main steamline isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1301 psig if a neutron flux scram is assumed. This results in a 74 psig margin to the code allowable overpressure limit of 1375 psig.

The analysis of the plant isolation transient (Load Rejection with bypass valve failure to open) assuming a turbine trip scram is presented in NEDO-24039-2 for Peach Bottom Unit 3. This analysis shows that the 11 safety/relief valves limit pressure at the safety valves to 25 psi below the setting of the safety valves. Therefore, the safety valves will not open.

The safety/relief valve settings satisfy the Code requirements that the lowest valve set point be at or below the vessel design pressure of 1250 psig. These settings are also sufficiently above the normal operating pressure range to prevent unnecessary cycling caused by minor transients.

The results of postulated transients where inherent safety/relief valve actuation is required are given in Section 14.0 of the Final Safety Analysis Report.

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome less than 75 psig.

LIMITING CONDITIONS FOR OPERATION

3.5.J Local LHGR (Cont'd)

If at any time during operation it is determined by normal surveillance that limiting value for LHGR is being exceeded, action shall be initiated within one (1) hour to restore LHGR to within prescribed limits. If the LHGR 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 LHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

During power operation, the MCPR for the applicable incremental cycle core average exposure and for each type of fuel shall be equal to or greater than the value given in Table 3.5-2 times kf, where kf is as shown in Figure 3.5.1.E. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within one (1) hour to restore MCPR to within prescribed limits. If the MCPR 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 MCPR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

MCPR shall be checked daily during reactor power operation at ≥25% rated thermal power.

Table 3.5-2

OPERATING LIMIT MCPR VALUES AS DETERMINED FROM INDICATED TRANSIENTS FOR VARIOUS CORE EXPOSURES

Fuel Type	MCPR Operating Limit For Incremental Cycle 3 Core Average Exposure	
	BOC to 2000 MWD/t Before EOC	2000 MWD/t before EOC To EOC
7x7	1.24 (RWE)	1.24 (RWE)
8x8 & 8x8R	1.24 (RWE or LH)	1.29 (LR)
8x8 PTA	1.24 (LH)	1.32 (LR)

RWE - Rod Withdrawal Error
LR - Load Rejection with failure of bypass valves to open
LH - Loss of 100°F Feedwater Heating

3.5.I BASES (Cont'd.)

A list of the significant plant parameters to the loss-of-coolant accident analysis is presented in Table 3.5-1.

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 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 ≥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.

Densification analyses for 8x8 fuel are presented in Section 5.2.3 of Reference 7.

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 1.07, and analyses of the abnormal operational transients presented in References 6, 7, & 9. 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.

3.5.K BASES (Cont'd.)

The limiting transients which determine the required steady state MCPR limits are given in Table 3.5-2. These transients yield the largest ΔCPR for each class of fuel. When added to the safety limit MCPR of 1.07, the required minimum operating limit MCPR's 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 at 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.

3.5.K BASES (Cont'd.)

A brief summary of the analytical method used to determine the nuclear characteristics is given in Section 3 of Reference 7.

Analysis of the abnormal operational transients is presented in Section 5.2 of Reference 7. Input data and operating conditions used in this analysis are shown in Table 5-3 of Reference 7 and Section 7 of Reference 9.

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 incore 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 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.

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. Supplemental Reload Licensing Submittal For Peach Bottom Atomic Power Station Unit 3 Reload No. 2, NEDO-24039-1, Supplement 1, December 1977.

7. General Electric Boiling Water Reactor Reload-2 Licensing Application For Peach Bottom Atomic Power Station Unit 3, NEDO-24039, August 1977.

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

9. Supplemental Reload Licensing Submittal For Peach Bottom Atomic Power Station Unit 3 Reload No. 2 Reanalysis Supplement, NEDO-24039-2, Supplement 2, May 1978.

3.6.D & 4.6.D BASES

Safety and Relief Valves

The safety/relief and safety valves are required to be operable above the pressure (122 psig) at which the core spray system is not designed to deliver full flow. The pressure relief system for each unit at the Peach Bottom APS has been sized to meet two design bases. First, the total capacity of the safety/relief and the safety valves has been established to meet the overpressure protection criteria of the ASME code. Second, the distribution of this required capacity between safety/relief valves and safety valves has been set to meet design basis 4.4.4.1 of subsection 4.4 of the FSAR which states that the nuclear system safety/relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME code requirements is presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report presented in Appendix K of the FSAR.

Eleven safety/relief valves and two safety valves have been installed on Peach Bottom Unit 3 with a total capacity of 79.51% of rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1301 psig if a neutron flux scram is assumed. This results in a 74 psig margin to the code allowable overpressure limit of 1375 psig.

To meet the power generation design basis, the total pressure relief system capacity of 79.51% has been divided into 65.96% safety/relief (11 valves) and 13.55% safety (2 valves). The analysis of the plant isolation transient (Load Rejection with bypass valve failure to open) assuming a turbine trip scram is presented in NEDO-24039-2. This analysis shows that the 11 safety/relief valves limit pressure at the safety valves to 25 psi below the setting of the safety valves. Therefore, the safety valves will not open.

Experience in safety/relief and safety valve operation shows that a testing of 50 per cent of the valves per year is adequate to detect failure or deteriorations. The safety/relief and safety valves are benchtested every second

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. DPR-56

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNIT NO. 3

DOCKET NO. 50-278

Introduction

By letter dated May 16, 1978,[1] Philadelphia Electric Company (the licensee) requested a change in Technical Specifications for the Peach Bottom Atomic Power Station, Unit No. 3 (PB-3). The reason for the request was to provide exposure-dependent minimum critical power ratio (MCPR) operating limits for the balance of cycle 3.

Evaluation

The present MCPR limits are as follows:

Fuel Type	MCPR operating limit	Limiting Transient
7x7	1.27	Fuel Loading Error (mislocated)
8x8	1.36	Fuel Loading Error (misoriented)
8x8R	1.36	Fuel Loading Error (misoriented)
PTA*	1.33	Load Rejection w/o Bypass

These limits were imposed by the staff[2] at the beginning of cycle 3 because of the results of the fuel loading error analysis[3]. Since then, the licensee has reanalysed[1] the fuel loading error using new methods which have recently been approved generically(5). The new analyses indicates that a misoriented bundle will result in a maximum ΔCPR of 0.12 instead of a ΔCPR of 0.29 which was previously reviewed by the staff. Thus, a mislocated bundle is no longer limiting since it will not result in a violation of the 1.07 safety limit MCPR. We find that the licensee's analysis of the fuel loading error is consistent with approved methods and the results are acceptable since the 1.07 safety limit MCPR is not violated.

*One Test Assembly

In addition, the other anticipated transients were reanalyzed using cycle-specific input data and two exposure intervals. In such analyses it is required to assume "worst cast" exposures (usually the higher exposure is more severe). This analysis of two exposure intervals, resulting in exposure-dependent limits, provides greater flexibility with no reduction in thermal safety margin. This procedure has since been approved generically[5].

During the previous review of cycle 3, the staff noted[2] that the Load Rejection without Bypass transient analysis had assumed[3] an initial thermal power level of 100% of rated, while the staff normally requires the assumption of at least 102% of rated thermal power (RTP) to account for calorimetric uncertainties. Although this transient was not limiting at the time, the removal of the fuel loading error as the limiting transient prompted re-evaluation of the Load Rejection without Bypass transient to ensure that adequate conservatism existed. The new analysis of this transient assumes 100% RTP for the second exposure interval only to demonstrate that the pressure margin to the safety valve setpoint was adequate; it did assume 104.5% RTP[6] for the ΔMCPR analyses for both intervals and is conservative relative to staff requirements for such analysis. Also, the older analysis was done both with and without assuming the trip of the recirculation M-G set drive motors. Because this specific reactor has drive motor trips on both high dome pressure and turbine control valve fast closure, the new analyses all assumed drive motor trip. The staff has reviewed these new inputs to the analyses, and finds them to be consistent with approved methods.

The new operating MCPR limits are based upon the most limiting transients as follows:

Fuel Type	Limit, BOC to 2000 MWD/t before EOC	Limiting Transient	Limit, 2000 MWD/t before to EOC	Limiting Transient
7x7	1.24	Rod Withdrawal Error	1.24	Rod Withdrawal Error
8x8	1.24	Rod Withdrawal Error	1.29	Turbine Trip or Load Rejection w/o Bypass
8x8R	1.24	Rod Withdrawal Error	1.29	
PTA	1.24	Rod Withdrawal Error	1.32	Load Rejection w/o Bypass

The staff finds these new limits acceptable based on the determination that the safety limit MCPR would not be violated. There is no change to the safety limit MCPR of 1.07 as previously reviewed and approved by the staff.

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) because the amendment 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 amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 14, 1978

References:

1. Letter, Philadelphia Electric Company (Eugene J. Bradley) to NRC (Director of Nuclear Reactor Regulation), dated May 16, 1978, enclosing "Application for Amendment of Facility Operating License DPR-56" and "Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit 3 Reload No. 2 Reanalysis Supplement," NEDO-24039-2.

2. Staff Safety Evaluation for Peach Bottom Nuclear Plant Unit No. 3, Reload 2, May 17, 1978.

3. "Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit 3 Reload No. 2," NEDO-24039-1.

4. Letter, NRC (Darrell G. Eisenhut) to General Electric Company (Ronald Engel) dated May 8, 1978.

5. Letter, NRC (Darrell G. Eisenhut) to General Electric Company (Richard Gridley), dated May 12, 1978.

6. Letter, Philadelphia Electric Company (Daltroff) to NRC (Director of Nuclear Reactor Regulation), dated June 2, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 42 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. 3. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications to incorporate exposure-dependent Minimum Critical Power Ratio (MCPR) operating limits for cycle 3 operation.

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 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 May 16, 1978, (2) Amendment No. 42 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 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 14 day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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