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Posted

Am-34 to DPR-44

April 4, 1977

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:

Tne Commission has issued the enclosed Amendment No. 34 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 March 7, 1977.

The amendment will allow a temporary change in the Technical Specifications to permit implementation of a testing program of reactor stability and transient response.

Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

George Lear

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

Enclosures:
1. Amendment No. 34
2. Safety Evaluation
3. FEDERAL REGISTER Notice

cc w/encl:
See next page

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. 34
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 March 7, 1977, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 34, 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: April 4, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 34

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace pages 9, 10, 11, 37, 38, 73 and 133b with the attached revised pages.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

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.

Objective

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

Specification:

A. Reactor Pressure ≥ 800 psia and Core Flow ≥ 10% of Rated

The existence of a minimum critical power ratio MCPR less then 1.06 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.

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.

Objective

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:

$S \leq 0.66W + 54\%$
* $S \leq 0.45W + 75\%$

where:

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

W = Loop recirculating flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr).

* Temporary change which applies only to the stability and transient tests authorized by Amendment No. 34 for EOC-2.

SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTING

2.1.A (cont'd)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified to the more limiting (lower) of the two values determined by the following:

a. $S \leq (0.66\ W + 54\%) \dfrac{2.63}{\text{MTPF for 7x7 fuel}}$

$*S \leq (0.45\ W + 75\%) \dfrac{2.63}{\text{MTPF for 7x7 fuel}}$

b. $S \leq (0.66\ W + 54\%) \dfrac{2.44}{\text{MTPF for 8x8 fuel}}$

$*S \leq (0.45\ W + 75\%) \dfrac{2.44}{\text{MTPF for 8x8 fuel}}$

MTPF = The value of the existing maximum total peaking factor

A = 2.63 for 7x7 fuel and 2.44 for 8x8 fuel. For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

2. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.

3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

* Temporary change which applies only to the stability and transient tests authorized by Amendment No. 34

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia)

When the reactor pressure is $\leq$ 800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 in. above the top of the normal active fuel zone.

*Temporary change which applies only to the stability and transient tests authorized by Amendment No. 34

**Technical Specification 2.1.D shall be deleted for the stability and testing authorized by Amendment No. 34

LIMITING SAFETY SYSTEM SETTING

B. APRM Rod Block Trip Setting

S_{RB} _ 0.66 + 42%

*S_{RB} _ 0.58 + 50%

where:

S_{RB} = Rod block setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr).

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified to the more limiting of the 2 values determined by the following:

1. S_{RB} _(0.66 W + 42%) $\frac{2.63}{\text{MTPF for 7x7 fuel}}$

   *S_{RB} _(0.58 W + 50%) $\frac{2.63}{\text{MTPF for 7x7 fuel}}$

2. S_{RB} _(0.66 W + 42%) $\frac{2.44}{\text{MTPF for 8x8 fuel}}$

   *S_{RB} _(0.58 W + 50%) $\frac{2.44}{\text{MTPF for 8x8 fuel}}$

MTPF = The value of the existing maximum total peaking factor

A = 2.63 for 7x7 fuel and 2.44 for 8x8 fuel.

C. Scram and isolation--reactor low water level $\geq$538 in. above vessel zero (0" on level instruments)

** D. Scram--turbine stop valve closure $\leq$10 percent

E. Scram--turbine control valve fast closure on loss of control oil pressure 500 P 850 psig.

Amendment No. ~~28~~, 34

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (7)	Startup	Run		
1	Mode Switch In Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
3	IRM High Flux	≤120/125 of Full Scale	X	X	(5)	8 Instrument Channels	A
3	IRM Inoperative		X	X	(5)	8 Instrument Channels	A
2	APRM High Flux	(.66W+54)(A/MTPF) (12)(13)			X	6 Instrument Channels	A or B
		*(0.45W + 75)(A/MTPF) (12)(13)			X	6 Instrument Channels	A or B
2	APRM Inoperative	(11)	X	X	X	6 Instrument Channels	A or B
2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
2	APRM High Flux in Startup	≤15% Power	X	X		6 Instrument Channels	A
2	High Reactor Pressure	≤1055 psig	X(9)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	≤2 psig	X(8)	X(8)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥0 in. Indicated Level	X	X	X	4 Instrument Channels	A

*Temporary change which applies only to the stability and transient tests authorized by Amendment No. 34

Amendment No. ~~23~~, 34

Tabl 3.1.1 (Cont'd.)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (7)	Startup	Run		
2	High Water Level in Scram Discharge Volume	≤50 Gallons	X(2)	X	X	4 Instrument Channels	A
2	Turbine Condenser Low Vacuum	≥23 in. Hg. Vacuum	X(3)	X(3)	X	4 Instrument Channels	A or C
2	Main Steam Line High Radiation	≤3 X Normal Full Power Background	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	8 Instrument Channels	A
2	Turbine Control Valve Fast Closure	500<P<850 psig Control Oil Pressure Between Fast Closure Solenoid and Disc Dump Valve			X(4)	4 Instrument Channels	A or D
*4	Turbine Stop Valve Closure	<10% Valve Closure			X(4)	8 Instrument Channels	A or D

* Deleted as temporary change which applies only to the stability and transient tests program authorized by Amendment No. 34

T[illegible] 3.2.C

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq [0.66\ W + 42]\ (\frac{A}{MTPF})$ (2)	6 Inst. Channels	(1)
* 2	APRM Upscale (Flow Biased)	$\leq [0.58\ W + 50]\ (\frac{A}{MTPF})$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Startup Mode)	$\leq$ 12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 2.5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq [0.66W + 41]\ (\frac{A}{MTPF})$ (2)	2 Inst. Channels	(1)
*1 (7)	Rod Block Monitor (Flow Biased)	$\leq [0.58\ W + 50]\ (\frac{A}{MTPF})$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (3)	$\geq$ 2.5 indicated on scale	8 Inst. Channels	(1)
3	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$ 108 indicated on scale	8 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)

*Temporary change which applies only to the stability and transient tests authorized by Amendment No. 34

LIMITING CONDITION OF 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, MCPR shall be $\geq$1.28 (1.31*) for 7x7 fuel and $\geq$1.31 (1.39*) for 8x8 fuel at rated power and flow. For core flows other than rated the MCPR shall be $\geq$1.28 (1.31*) times k_f for 7x7 fuel and $\geq$1.31 (1.39*) times k_f for 8x8 fuel where k_f 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 prescribes 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 prescribed limits.

*Temporary change which applies only to the stability and transient tests authorized by Amendment No. 34

SURVEILLANCE REQUIREMENT

4.5.K. Minimum Critical Power Ratio (MCPR)

MCPR shall be checked daily during reactor power operation at $\geq$ 25% rated thermal power.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 34 TO FACILITY LICENSE NO. DPR-44

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNIT NO. 2

DOCKET NO. 50-277

Introduction

By application for license amendment dated March 7, 1977, [1,2] the Philadelphia Electric Company (PECO) requested temporary changes to Peach Bottom Atomic Power Station, Unit No. 2 (PB-2) Technical Specifications. These requested changes would provide revised safety limit settings for a proposed stability and transient test program. During the course of Staff review, it was determined that additional information was required. The March 7, 1977 application was supplemented by letter dated March 31, 1977[3] in response to NRC's request for additional information [4]. These tests will be used for the verification and future development of the stability and transient analytical methods.

Evaluation

The proposed stability and transient test program consists of two independent reactor test series. The proposed stability test series will be at four different reactor operating conditions. At each of the four reactor operating conditions, small pressure perturbations will be input into the core power-to-void dynamic feedback loop and the associated system response will be monitored. In order to perform these proposed stability tests, the APRM scram and rod block lines must be adjusted and Technical Specification changes to that effect have been proposed (Technical Specification Sections 2.1.A, 2.1.B, Tables 3.1.1 and 3.2.C).

The proposed transient test series will consist of three manual turbine trips with bypass. In order to ensure appropriate test conditions, i.e., small measurable neutron power rise, the anticipatory turbine stop valve closure scram must be disabled. To ensure adequate safety margin, which is equivalent to that previously demonstrated in the transient and accident analyses for cycle 2 operation, for this change compensatory Technical Specification changes have been proposed to the MCPR limits (Technical Specification 3.5.K).

A. Stability Tests

The stability tests are to be conducted as closely as possible to the four test operating conditions as specified in Figure 1 of Reference 1. These test operating conditions were selected to represent a locus or points about the design reference condition for the Peach Bottom-2 core stability. The results of the tests for reactor core stability characteristics in this range of interest will be compared to the current calculational predictions and will be used in the future refinement of stability calculational techniques.

The tests will be performed in order of decreasing stability margin. The data from each preceeding test will be extrapolated to the next test condition for the predicted stability margin. Based on this calculational technique, the licensee has proposed an acceptable test performance limit[3]. This test performance limit allows for reasonable calculational uncertainties and provides an acceptable stability margin for experimentally controlled conditions.

For each test, a series of small setpoint perturbations in the reactor pressure control system, i.e., Electric-Hydraulic Control system (EHC), will be monitored and the associated system response will be measured. These pressure regulator setpoint step inputs will initiate action of the turbine control valve or the steam bypass valves. The turbine control or the bypass valves would open or close to a degree to conform to the new pressure regulator setpoint. The transient response to this change, i.e., neutron flux, flow rate, pressure, temperature, etc., will be monitored and related to system stability characteristics.

The pressure setpoint disturbances will be a combination of periodic and random step changes of about 10 psi amplitude for approximately 15 minutes at each test point. Then, about 10 minutes of steady-state noise data will also be recorded at each test point. The data from these test points will be recorded by a digital data acquisition system. The measurement parameters and equipment are discussed in Section 2.1 and Table 2.1 of reference 2 and do not interface with reactor safety systems. This testing technique does not affect the probabilities or consequences of the transients and accidents previously analyzed in the safety analyses and therefore, is acceptable.

The changes to Technical Specification would increase the APRM rod block and high flux scram lines for test conditions. The APRM rod block line is required in order to attain test condition PT4 of Figure 1, Reference 1. The APRM rod block line serves as protection against the rod withdrawal error (RWE) transient. The most restrictive RWE has been analyzed in the Final Safety Analysis Report (Chapter IX, Reference 5) at "design" power conditions. The results of the RWE from the proposed test conditions have been shown by analysis to be less severe than the RWE of the FSAR [3].

The change to APRM high flux scram line is required in order to avoid spurious trip due to test perturbations. "The APRM scram line was originally set to increase the margin for low flow transients; however, no credit for this scram was taken in the FSAR..." (Chapter 14, Reference 5). Therefore, temporarily adjusting the scram line would not significantly decrease the safety margin, and is, therefore, acceptable.

The Technical Specification changes will be implemented after reaching the first stability test point. Upon completion of these tests, the reactor will be adjusted to the rated power-flow control line at minimum pump speed and the APRM rod block and high flux scrams will be set to their previously approved cycle 2 values.

The licensee has also discussed the effect of the most limiting stability situation. For a self-sustained oscillation of a system parameter at its maximum possible amplitude, e.g., the APRM flux scram limit minus flow control line flux, the licensee has provided analyses which show that safety limits for the thermal hydraulic (MCPR $\geq$1.06) and mechanical (plastic strain $\leq$1%) limits would not be violated(3). Any other flux oscillation stability transient would be self-damping or terminated by reactor protection scrams. Therefore, the operation of the proposed test condition would not affect reactor safety. The staff agrees with this evaluation.

Based on the above discussion the staff finds the proposed temporary change to Technical Specification for test performance to be acceptable.

B. Transient Tests

The transient test program is to consist of as many as three rapid core pressurization transients. The test will be conducted at three different power levels as indicated in Table 2 of Reference 1 and at rated core flow. The tests are to simulate the turbine trip with bypass at these conditions. For these tests the steam turbine will be tripped from the initial conditions and system response will be measured. These test data will be used in the verification and development of reactor transient analysis techniques.

The licensee has proposed the following changes for the transient tests:

1) Disconnect the reactor scram function which actuates on closure of the main steam stop valves.

2) Modify the Technical Specification on operating limit MCPR's to compensate for effect of the elimination of the scram on stop valve closure.

The main steam stop valve scram function anticipate high flux due to loss of heat sink to the turbine and subsequent void collapse due to pressure increase. It is necessary to eliminate this scram function in order to obtain sufficient core responses for the verification and development purpose of these tests. With this modification, the plant operating limits must also be modified in order to provide sufficient protection from anticipated transients which is equivalent to that previously calculated for cycle 2 operation.

The elimination of this scram function affects the turbine trip without bypass transients. The turbine trip without bypass is the most restrictive transient for both configuration, i.e., with and without turbine stop valve closure scram. A safety analyses for this most restrictive design basis transient (turbine trip without bypass) has been performed (2) with initial conditions at the most limiting test point (TT3) of Table 2 of References 1. This analysis found ΔMCPR values in excess of those previously used. In this new analysis, the reactor scram is initiated slightly later in the transient by a high flux signal, rather than by a (disabled) stop valve closure scram signal. Thus, by the proposed modification of the operating limit MCPR (Table 3 of reference 1), the change provides an acceptable compensation for the disabling of the turbine steam stop valve closure scram. This change in operating limit MCPR ensures that the occurrence of the "licensing basis transient" during the tests will not result in a violation of the plant safety limit MCPR and, thereby, provide the same safety margin to departure from nucleate boiling as was previously established for cycle 2 operation.

This Technical Specification change will be implemented after the reactor has been brought to the first test condition. The interim operating MCPR limit would be imposed and the anticipatory trip scram function disabled at this lowest initial power test condition. Before reactor restart after the tests, the stop valve closure scram will be reinstated to its previous normal operating condition and the operating MCPR limits will be restored to their original cycle 2 values.

Test predictions have been presented in section 3.3 of reference 2. These test predictions show that no safety limit will be violated by the proposed test performance. The tests will be conducted in order of increasing power. After each test the results will be examined and a prediction of the limiting transient for the next test condition will be made with the aid of the previous test results. If this adjusted licensing basis transient does not violate plant safety limits, the test will proceed. If the adjusted analysis predicts safety limit violation, the transient test program will not be performed.

The APRM flux scram line will be lowered closer to the initial operating power condition prior to the tests in order to minimize steam relief valve venting during the latter part of the test transient. This is in the conservative direction and corresponds to an "earlier" trip. Therefore, this is acceptable on the basis that the ARPM flux scram line will be less than or equal to the previously approved value per cycle 2 Technical Specifications.

Since the test acceptance criteria have been satisfied and the Technical Specification for operating limit MCPR has been changed to provide a compensating condition, we have assurance that the health and safety of the public will not be endangered by operation at the proposed test conditions.

Environmental Considerations

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 change does not involve a significant increase in the probability or consequences of accidents previously considered and 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: April 4, 1977

REFERENCES

1. Letter from Eugene J. Bradley (PECO) to Bernard C. Rusche, (NRC) dated March 7, 1977.

2. "Proposed Peach Bottom Atomic Power Station Unit No. 2 Stability and Transient Test Program," General Electric Company, Boiling Water Reactor Systems Department, San Jose, California 95125, NEDO-24003, Class I,

3. Letter from Edward G. Bauer, Jr. (PECO) to George Lear (NRC) dated March 31, 1977.

4. Letter from G. Lear (NRC) to Edward G. Bauer, Jr, (PECO) dated March 31, 1977.

5. Final Safety Analysis Report, Peach Bottom Atomic Power Station Unit Nos. 2 and 3, Philadelphia Electric Company, Docket No. 50-278.

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. 34 to Facility Operating License No. DPR-44 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 consists of temporary changes to the Technical Specifications which will allow implementation of a testing program of reactor stability and transient response. During the tests compensatory changes have been made that will assure that reactor safety limits will be the same as those established by transient and accident analyses for cycle 2 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 March 7, 1977, (2) Amendment No. 34 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 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 4th day of April 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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