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Dockets Nos. 50-277
and 50-278

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 Amendments Nos. 7 and 4 to Facility Operating Licenses Nos. DPR-44 and DPR-56, respectively, for the Peach Bottom Atomic Power Station, Units 2 and 3. These amendments include Changes Nos. 7 and 4 to the Technical Specifications and are in response to your request of August 29, 1974.

The amendments delete the provisions in the Technical Specifications which require that the maximum reactivity worth of any operable control rod be less than 1.25% when the reactor is operated above 30% rated power.

A copy of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

/s/

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

Enclosures:
1. Amendments Nos. 7 and 4
2. Safety Evaluation
3. Federal Register Notice

cc: w/enclosure
See next page

PECo AMDTS 6 and 4 Changes 7 and 4 DPR-44 and DPR-56	ORB #3	ORB#3	TR:AD	ORB#3	OELD	L:AD/ORs	D:DLONRR
OFFICE ➤							
SURNAME ➤	SATeets:kmf	DHJaffe:kmf	VStello	GLear	Scinto	KRGoller	AGiambusso
DATE ➤	1/31/75	1-31-75	2/18	2/18	2/26	2/28/75	3/27/75

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

cc:- w/enclosures

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OFFICE ➤						
SURNAME ➤						
DATE ➤						

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 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.6
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 the Atlantic City Electric Company ("the licensees") dated August 29, 1974, 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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility 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, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 7."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 7 to the
Technical Specifications

Date of Issuance: FEB 2 8 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 6

CHANGE NO. 4 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace pages 101-102 and 109-110 with the attached revised pages. (No change has been made on page 101.)

Add the attached page 110a.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B (cont'd.)	4.3.B (cont'd.)
B. Control Rods	
1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.	b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position. c. During each refueling outage and after control rod maintenance, observe that the drive does not go to the overtravel position.
2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.
3. a. Whenever the reactor is in the startup or run modes below 30% rated power the Rod Sequence Control System shall be operable, that is no position switches shall be bypassed except as permitted in 3.3.A.2d, except during shut down margin testing. b. Whenever the reactor is in the startup or run modes below 25% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.	3. Prior to the start of control rod withdrawal towards criticality, and prior to attaining 25% rated power during rod insertion at shutdown, the capability of the Rod Worth Minimizer and Rod Sequence Control System to properly fulfill their functions shall be verified by the following checks: a. The capability of the Rod Sequence Control System to properly fulfill its function shall be verified by attempting to select and move a rod in each of the out-of-sequence groups.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B (cont'd.)	4.3.B (cont'd.)
7	b. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks: 1. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified. 2. The RWM computer on line diagnostic test shall be successfully performed. 3. Prior to the start of control rod withdrawal only, proper annunication of the selection error of at least one out-of-sequence control rod in a fully inserted group shall be verified. 4. The rod block function of the RWM shall be verified by withdrawing the first rod during start-up only as an out-of-sequence control rod no more than to the block point.
d. If Specifications 3.3.B.3.a through c cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 25% rated power, it shall be brought to a shutdown condition immediately.	c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified and recorded.
e. Prior to the end of the first fuel cycle, analyses must be provided to the Directorate of Licensing to show that the Rod Sequence Control System will limit the control rod drop accident to a peak fuel enthalpy less than 280 calories per gram and doses to less than 10 CFR Part 100 guideline values for fuel loadings planned after the first fuel cycle.	

3.3 and 4.3 BASES (Cont'd.)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection 3.5.2 of the FSAR and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. All patterns associated with these sequences have the characteristic that, assuming the worst single deviation from the sequence, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in the average enthalpy of any pellet exceeding 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Sections 3.6.6, 14.6.2 and 7.16.3.3 of the FSAR and NEDO-10527 and supplements thereto. 7

In performing the function described above, the RWM and RSCS are not needed to impose any restrictions at core power levels in excess of 20 percent of rated power; however, Technical Specifications require the use of the RWM below 25% rated power and the RSCS below 30% of rated power. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at a power level greater than 20 percent, regardless of the rod pattern. This is true for all normal and abnormal patterns, including those which maximize individual control rod worth. 7

At power levels below 20 percent of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod

3.3 and 4.3 BASES (Cont'd)

sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., they limit operator deviations from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls. The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At power levels below 20 percent of rated these devices force adherence to acceptable rod patterns. Above 20 percent of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20 percent of rated power are imposed by power distribution requirements as defined in Section 3.5/4.5 of the Technical Specifications. 7

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3.3 and 4.3 BASES (Cont'd)

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCHFR = 1.9 or LHGR = 18.5kW/ft). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the station superintendent.

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

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

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company and the Atlantic City Electric Company ("the licensees") dated August 29, 1974, 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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility 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, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 4."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 4 to the
Technical Specifications

Date of Issuance: FEB 2 8 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 4

CHANGE NO. 4 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace pages 101-102 and 109-110 with the attached revised pages. (No change has been made on page 101.)

Add the attached page 110a.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B (cont'd.)	4.3.B (cont'd.)
B. Control Rods	
1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.	b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position. c. During each refueling outage and after control rod maintenance, observe that the drive does not go to the overtravel position.
2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.
3. a. Whenever the reactor is in the startup or run modes below 30% rated power the Rod Sequence Control System shall be operable, that is no position switches shall be bypassed except as permitted in 3.3.A.2d, except during shut down margin testing. b. Whenever the reactor is in the startup or run modes below 25% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.	3. Prior to the start of control rod withdrawal towards criticality, and prior to attaining 25% rated power during rod insertion at shutdown, the capability of the Rod Worth Minimizer and Rod Sequence Control System to properly fulfill their functions shall be verified by the following checks: a. The capability of the Rod Sequence Control System to properly fulfill its function shall be verified by attempting to select and move a rod in each of the out-of-sequence groups.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B (cont'd.)	4.3.B (cont'd.)
4	b. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks: 1. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified. 2. The RWM computer on line diagnostic test shall be successfully performed. 3. Prior to the start of control rod withdrawal only, proper annunication of the selection error of at least one out-of-sequence control rod in a fully inserted group shall be verified. 4. The rod block function of the RWM shall be verified by withdrawing the first rod during start-up only as an out-of-sequence control rod no more than to the block point.
d. If Specifications 3.3.B.3.a through c cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 25% rated power, it shall be brought to a shutdown condition immediately.	c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified and recorded.
e. Prior to the end of the first fuel cycle, analyses must be provided to the Directorate of Licensing to show that the Rod Sequence Control System will limit the control rod drop accident to a peak fuel enthalpy less than 280 calories per gram and doses to less than 10 CFR Part 100 guideline values for fuel loadings planned after the first fuel cycle.	

3.3 and 4.3 BASES (Cont'd.)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection 3.5.2 of the FSAR and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. All patterns associated with these sequences have the characteristic that, assuming the worst single deviation from the sequence, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in the average enthalpy of any pellet exceeding 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Sections 3.6.6, 14.6.2 and 7.16.3.3 of the FSAR and NEDO-10527 and supplements thereto. 4

In performing the function described above, the RWM and RSCS are not needed to impose any restrictions at core power levels in excess of 20 percent of rated power; however, Technical Specifications require the use of the RWM below 25% rated power and the RSCS below 30% of rated power. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at a power level greater than 20 percent, regardless of the rod pattern. This is true for all normal and abnormal patterns, including those which maximize individual control rod worth. 4

At power levels below 20 percent of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod 4

3.3 and 4.3 BASES (Cont'd)

sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., they limit operator deviations from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls. The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At power levels below 20 percent of rated these devices force adherence to acceptable rod patterns. Above 20 percent of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20 percent of rated power are imposed by power distribution requirements as defined in Section 3.5/4.5 of the Technical Specifications. 4

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3.3 and 4.3 BASES (Cont'd)

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCHFR = 1.9 or LHGR = 18.5kW/ft). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the station superintendent.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 6 AND 4 TO LICENSES

NOS. DPR-44 AND DPR-56

(CHANGES NOS. 7 AND 4 TO THE TECHNICAL SPECIFICATIONS)

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

Introduction

On August 29, 1974, the Philadelphia Electric Company (PECo) submitted an application for amendment of the Facility Operating Licenses that would delete Technical Specifications 3.3.B.3.c and modify Section 3.3.B.3 and 4.3.B.3 Bases. This change would remove the limit of 1.25%Δρ placed on the maximum worth of any control rod when the reactor is operated above 30% of rated power.

Discussion

At the present time, Technical Specification 3.3.B.3.c is stated as follows: "When the reactor is above 30% rated power, control rod patterns and the withdrawal or insertion sequences shall be established such that the maximum worth of any operable control rod, including an allowance for a selection error, is less than 1.25%Δρ." The intent of this Technical Specification was to insure that any transient (power excursion) resulting from a dropped control rod, while the reactor operated above 30% of rated power, would not exceed the fuel design enthalpy limit. This design limit, for General Electric (GE) BWR fuel, is an enthalpy of 280 cal/gm; no fuel dispersion is expected for fuel enthalpies below this value.

GE has presented a report, NEDO-10527, Supplement 1, which indicates the effect on the fuel design enthalpy limit of transients caused by control rod drop accidents in large BWR's. This report indicates that above 5% of rated power, rod drop accidents involving the worst operator error will always result in peak fuel enthalpies less than 280 cal/gm at a core average burn-up of 6500 MWD/T. This particular fuel exposure was chosen since it gives the highest worth control rod its maximum value. Based on our review of this report, we concluded that NEDO-10527, Supplement 1, provides sufficient assurance that core damage would not result if a control rod were to be dropped, under the most adverse conditions while the reactor operates above 30% of rated power. Accordingly, Technical

Specification 3.3.B.3.c should be deleted and the corresponding Bases modified.

Another conclusion of NEDO-10527, Supplement 1, is that for low power levels (below 5% of rated power), transients caused by dropped control rods can exceed the fuel design enthalpy limit of 280 cal/gm. For this reason Peach Bottom Units 2 and 3 are provided with independent and redundant systems to prevent operator error in the selection of control rod movement patterns. These systems, referred to as the Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS), are required by Technical Specifications to be operable below 25% and 30% rated power, respectively. By constraining the operator to the use of preselected (in-sequence) control rod movement patterns, it is not possible to select a control rod combination which would inadvertently maximize the reactivity worth of a single control rod. NEDO-10527, Supplement 1, concludes that rod drop accidents involving in-sequence control rods will always result in peak fuel enthalpies less than 280 cal/gm. Thus, these systems (RWM and RSCS) provide sufficient protection against violation of the fuel design limit for energy (enthalpy) deposition, at low power, due to control rod drop transients. We conclude from the above that, upon deletion of Technical Specification 3.3.B.3.c, sufficient protection remains in the Technical Specifications and inherent design features to provide protection against the occurrence of a dropped control rod and the associated power transient at any permissible reactor power level.

The specification of acceptable rod patterns, with regard to dropped control rod transients, is sensitive to both power level and core loading. It is anticipated that the licensee will resubmit a dropped control rod analysis, prior to each core reload, to demonstrate that the control rod patterns to be used in the RSCS and RWM are acceptable with regard to dropped control rod transients. Technical Specification 3.3.B.3.e. requires that such an analysis be performed for the RSCS prior to the second cycle.

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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

FEB 2 8 1975

BIBLIOGRAPHY

1) Stirm, R, Poon, C., Young, R., "Rod Drop Accident Analysis for Large BWR's", NEDO-10527, Supplement1, General Electric, July, 1972.

U. S. NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

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

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 6 and 4 to Facility Operating Licenses Nos. DPR-44 and DPR-56, respectively issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company and the Atlantic City Electric Company which revised Technical Specifications for operation of the Peach Bottom Atomic Power Station Units 2 and 3, located in Peach Bottom, York County, Pennsylvania. The amendments are effective as of date of issuance.

The amendments delete the provisions in the Technical Specifications which require that the maximum worth of any operable control rod be less than 1.25% when the reactor is operated above 30% rated power.

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 in connection with this action was published in the Federal Register on November 6, 1974 (39 F.R. 39311). No request for a hearing or petition for leave to intervene was filed following the notice of the proposed action.

For further details with respect to this action, see (1) the application for amendment dated August 29, 1974, (2) Amendments Nos. 6 and 4 to Licenses Nos. DPR-44 and DPR-56, with Changes Nos. 7 and 4, respectively, 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.

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 Reactor Licensing.

Dated at Bethesda, Maryland, this 28thday of February, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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