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Docket 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. 43 and 43 to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Units Nos. 2 and 3. The amendments revise the Technical Specifications in response to your request of April 14, 1978.

These amendments revise the Technical Specifications to reduce the power level above which Rod Sequence Control System is no longer necessary. The reduction is from 30% of rated power to 21%.

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

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 43 to DPR-44
2. Amendment No. 43 to DPR-56
3. Safety Evaluation
4. Notice

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE➤ | ORB#3 | ORB#3 | OELD | ORB#3 | | |
| SURNAME➤ | SSheppard | DVerrelli:ar | JMC | TIppolito | | |
| DATE➤ | 6/20/78 | 6/20/78 | 6/21/78 | 6/22/78 | | |

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. 43
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated April 14, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 43, 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

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: June 23, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

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

| Remove | Insert |
|---|---|
| 99 | 99 |
| 99a | - |
| 100 | 100* |
| 100a | - |
| 101 | 101 |
| 102 | 102 |
| - | 102a |
| 109 | 109 |
| 240e | 240e |

*No changes on this page

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive condition during the the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.

2. Reactivity margin - inoperable control rods

a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable.

If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability:

Applies to the surveillance requirements of the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38%$\Delta k/k$ that the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest operable control rod fully withdrawn and all other operable rods fully inserted.

2. Reactivity margin - inoperable control rods

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 30%(*) power. This test shall be performed at least once per 24 hours in the event power operation above 30%(*) is continuing with three or more inoperable control rods or in the event power operation above 30%(*) is continuing with one fully

*After installation of improved instrumentation authorized by Amendment No. 43, 21% power limit applies.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.3.A. Reactivity Limitations (Cont'd) | 4.3.A Reactivity Limitations (Cont'd) |
| failure is not due to a failed control rod drive mechanism collet housing. | or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than 3 and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod. |
| b. The control rod directional control valves for inoperable control rods shall be disarmed electrically and the control rods shall be in such positions that Specification 3.3.A.1 is met. | b. A second licensed operator shall verify the conformance to Specification 3.3.A.2d before a rod may be bypassed in the Rod Sequence Control System. |
| c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically. | |
| d. Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System and considered operable if the actual rod position is known. These rods must be moved in sequence to their correct positions (full in on insertion or full out on withdrawal.) | |
| e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable. | |

LIMITING CONDITIONS FOR OPERATION

3.3.A. Reactivity Limitations (Cont'd)

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5 x 5 array may be inoperable (at least 4 operable control rods must must separate any 2 inoperable ones). If this Specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a cold shutdown condition within 24 hours.

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.

*After installation of improved instrumentation authorized by Amendment No. 43, 21% power limit applies.

SURVEILLANCE REQUIREMENTS

4.3.A Reactivity Limitations (Cont'd)

B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

   a. When a rod is withdrawn the first time after each refueling outage or after after maintenance, observe discernible response of the nuclear instrumentation and rod position indication for the "full-in" and "full-out" position. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 30%(*) power shall be performed to verify instrumentation response.

   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.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.3.B Control Rods (Cont'd) | 4.3.B Control Rods (Cont'd) |
| | 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 shutdown margin testing. | 3. a. The "sequence" mode of RSCS shall be demonstrated to be operable by attempting to select and move a rod in each of the out-of-sequence groups.<br>1. Prior to the start of control rod withdrawal for a reactor start-up.<br>2. As soon as the "sequence" mode of RSCS is automatically initiated during rod insertion when reducing power.<br>The "group notch" mode of RSCS shall be demonstrated to be operable by attempting to move a control rod more than one notch in the first programmed group;<br>1. After reaching 50% rod density on a reactor startup.<br>2. As soon as the "group notch" mode of RSCS is automatically initiated during rod insertion when reducing power. |

*After installation of improved instrumentation authorized by Amendment No. 43, 21% power limit applies.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.3.B Control Rods (Cont'd) | 4.3.B Control Rods (Cont'd) |
| 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. | b. Prior to the start of control rod withdrawal towards criticality and prior to attaining 25% of rated power during rod insertion at shutdown, the Rod Worth Minimizer (RWM) shall be demonstrated to be operable by the following checks:<br>1. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.<br>2. The RWM computer on line diagnostic test shall be successfully performed.<br>3. Prior to the start of control rod withdrawal only, proper annunciation of the selection error of at least one out-of-sequence control rod in a fully inserted group shall be verified.<br>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. |
| c. If Specifications 3.3.B.3a or b 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. |

*After installation of improved instrumentation authorized by Amendment No. 43, 21% power limit applies.

3.3.B and 4.3.B <u>BASES</u> (Cont'd.)

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. The Technical Specifications, for conservatism, require the use of the RWM below 25% rated power. The 30%(*) power operability requirement for the RSCS is based on the evaluation of the instrumentation, pressure correlation, and other errors associated with determining the reactor thermal 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.

Up to 50% rod density (either sequence A or B control rods fully withdrawn and the other sequence fully inserted), the sequence mode of the RSCS restricts the maximum positive reactivity which can be added to the core due to a dropped control rod by control rod selection. Between 50% rod density and 30%(*) of rated power, the group notch mode of the RSCS restricts the reactivity worth by requiring movement of control rods such that rods assigned to each notch group are kept within one notch of each other.

The Rod Worth Minimizer and the sequence mode of the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted and the group notch mode of RSCS requires notch movement of rods; i.e., the systems limit operator deviations from planned control rod movement. They serve as a backup to procedural control of control rod movement, 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. 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.

*After installation of improved instrumentation authorized by Amendment No. 43, 21% power limit applies.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| With one fire pump or logic inoperable, restore the equipment to an operable status within 7 days, or in lieu of any other report required by Specification 6.9.2, submit a Special Report to the Commission pursuant to Specification 6.9.3 within 31 days outlining the cause of the malfunction and the plans for restoring the equipment to an operable status. Reactor startup and/or continued reactor operation is permissible. | 2. With one fire pump inoperable, the remaining fire pump shall be demonstrated to be operable immediately and at least every 72 hours thereafter until the inoperable pump is restored to an operable status. |
| 3. With two fire pumps inoperable,<br>a. establish a back-up water supply within 24 hours,<br>b. notify the Commission pursuant to Specification 6.9.2.a within 24 hours, by telephone and in writing no later than the first working day following the event. Submit a report within 14 days outlining the actions taken and the plans and schedule for restoring the equipment to an operable status, and<br>c. restore the equipment to an operable status within 14 days.<br>d. If a. above cannot be fulfilled, place the reactor in Hot Standby within the next six hours and in Cold Shutdown within the following thirty hours. | 3. None |
| 4. Except as specified in 3.14.A.6 below, the fire hose stations serving the following structures shall be operable:<br>a. Reactor Buildings<br>b. Radwaste Building | 4. The fire hose station inspections shall be performed as follows:<br>a. Visual inspection of hose station equipment availability - once every 31 days.<br>b. Hose and gasket inspection - once every 18 months. |

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

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

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated April 14, 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. 43, 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

TAIppolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: June 23, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 43

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 | Insert |
|---|---|
| 99 | 99 |
| 99a | - |
| 100 | 100* |
| 101 | 101 |
| 102 | 102 |
| - | 102a |
| 109 | 109 |
| 240e | 240e |

*No change to this page

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive condition during the the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.

2. Reactivity margin - inoperable control rods

a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable.

If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability:

Applies to the surveillance requirements of the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38%$\Delta k/k$ that the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest operable control rod fully withdrawn and all other operable rods fully inserted.

2. Reactivity margin - inoperable control rods

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 21% power. This test shall be performed at least once per 24 hours in the event power operation above 21% is continuing with three or more inoperable control rods or in the event power operation above 21% is continuing with one fully

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.3.A. Reactivity Limitations (Cont'd) | 4.3.A Reactivity Limitations (Cont'd) |
| failure is not due to a failed control rod drive mechanism collet housing. | or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than 3 and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod. |
| b. The control rod directional control valves for inoperable control rods shall be disarmed electrically and the control rods shall be in such positions that Specification 3.3.A.1 is met. | b. A second licensed operator shall verify the conformance to Specification 3.3.A.2d before a rod may be bypassed in the Rod Sequence Control System. |
| c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically. | |
| d. Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System and considered operable if the actual rod position is known. These rods must be moved in sequence to their correct positions (full in on insertion or full out on withdrawal.) | |
| e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable. | |

LIMITING CONDITIONS FOR OPERATION

3.3.A. Reactivity Limitations (Cont'd)

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5 x 5 array may be inoperable (at least 4 operable control rods must must separate any 2 inoperable ones). If this Specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a cold shutdown condition within 24 hours.

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.

SURVEILLANCE REQUIREMENTS

4.3.A Reactivity Limitations (Cont'd)

B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

   a. When a rod is withdrawn the first time after each refueling outage or after after maintenance, observe discernible response of the nuclear instrumentation and rod position indication for the "full-in" and "full-out" position. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 21% power shall be performed to verify instrumentation response.

   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.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.3.B Control Rods (Cont'd) | 4.3.B Control Rods (Cont'd) |
| | 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 21% 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 shutdown margin testing. | 3. a. The "sequence" mode of RSCS shall be demonstrated to be operable by attempting to select and move a rod in each of the out-of-sequence groups.<br>1. Prior to the start of control rod withdrawal for a reactor start-up.<br>2. As soon as the "sequence" mode of RSCS is automatically initiated during rod insertion when reducing power.<br>The "group notch" mode of RSCS shall be demonstrated to be operable by attempting to move a control rod more than one notch in the first programmed group;<br>1. After reaching 50% rod density on a reactor startup.<br>2. As soon as the "group notch" mode of RSCS is automatically initiated during rod insertion when reducing power. |

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.3.B Control Rods (Cont'd) | 4.3.B Control Rods (Cont'd) |
| 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. | b. Prior to the start of control rod withdrawal towards criticality and prior to attaining 25% of rated power during rod insertion at shutdown, the Rod Worth Minimizer (RWM) shall be demonstrated to be operable by the following checks:<br>1. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.<br>2. The RWM computer on line diagnostic test shall be successfully performed.<br>3. Prior to the start of control rod withdrawal only, proper annunciation of the selection error of at least one out-of-sequence control rod in a fully inserted group shall be verified.<br>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. |
| c. If Specifications 3.3.B.3a or b cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 21% 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. |

3.3.B and 4.3.B <u>BASES</u> (Cont'd.)

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. The Technical Specifications, for conservatism, require the use of the RWM below 25% rated power. The 21% power operability requirement for the RSCS is based on the evaluation of the instrumentation, pressure correlation, and other errors associated with determining the reactor thermal 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.

Up to 50% rod density (either sequence A or B control rods fully withdrawn and the other sequence fully inserted), the sequence mode of the RSCS restricts the maximum positive reactivity which can be added to the core due to a dropped control rod by control rod selection. Between 50% rod density and 21% of rated power, the group notch mode of the RSCS restricts the reactivity worth by requiring movement of control rods such that rods assigned to each notch group are kept within one notch of each other.

The Rod Worth Minimizer and the sequence mode of the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted and the group notch mode of RSCS requires notch movement of rods; i.e., the systems limit operator deviations from planned control rod movement. They serve as a backup to procedural control of control rod movement, 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. 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.

LIMITING CONDITIONS FOR OPERATION

With one fire pump or logic inoperable, restore the equipment to an operable status within 7 days, or in lieu of any other report required by Specification 6.9.2, submit a Special Report to the Commission pursuant to Specification 6.9.3 within 31 days outlining the cause of the malfunction and the plans for restoring the equipment to an operable status. Reactor startup and/or continued reactor operation is permissible.

3. With two fire pumps inoperable,

   a. establish a back-up water supply within 24 hours,

   b. notify the Commission pursuant to Specification 6.9.2.a within 24 hours, by telephone and in writing no later than the first working day following the event. Submit a report within 14 days outlining the actions taken and the plans and schedule for restoring the equipment to an operable status, and

   c. restore the equipment to an operable status within 14 days.

   d. If a. above cannot be fulfilled, place the reactor in Hot Standby within the next six hours and in Cold Shutdown within the following thirty hours.

4. Except as specified in 3.14.A.6 below, the fire hose stations serving the following structures shall be operable:

   a. Reactor Buildings
   b. Radwaste Building

SURVEILLANCE REQUIREMENTS

2. With one fire pump inoperable, the remaining fire pump shall be demonstrated to be operable immediately and at least every 72 hours thereafter until the inoperable pump is restored to an operable status.

3. None

4. The fire hose station inspections shall be performed as follows:

   a. Visual inspection of hose station equipment availability - once every 31 days.

   b. Hose and gasket inspection - once every 18 months.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NOS. 43 AND 43 TO FACILITY LICENSE NOS. DPR-44 AND DPR-56

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

PEACH BOTTOM ATOMIC POWER STATION UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

## I. Introduction

By letter dated April 14, 1978 (Reference 1), the Philadelphia Electric Company, the licensee, applied for an amendment of Facility Operating License for Peach Bottom Atomic Power Station (PBAPS) Units Nos. 2 and 3. This letter requested changes to the PBAPS Technical Specifications which reflect a proposed modification to the Rod Sequence Control System (RSCS). These changes reduce the power level above which RSCS operation is no longer necessary. The reduction would be from 30% of rated power to 21%.

## II. Evaluation

RSCS is designed to limit the worth of reactor control rods. This limitation mitigates the consequences of any potential control rod drop accident (RDA). A RDA would result in a rapid insertion of reactivity into the reactor core and subsequent localized power excursion. A RDA is a design basis accident for the PBAPS Units Nos. 2 and 3, and is required to meet the peak fuel enthalpy criterion of $\leq$280 calories/gram. The $\leq$280 calories/gram criterion was established below the enthalpy at which rapid fuel dispersal could occur and on that basis has been previously found acceptable for the RDA evaluation (Reference 2). Thus, the RSCS has been designed to limit control rod worths such that the peak fuel enthalpy for the RDA will not exceed 280 calories/gram.

The modification to the RSCS consists of the replacement of the existing turbine first stage pressure switches with instrumentation loops of pressure transmitters and trip units. (The turbine first stage pressure can be correlated to core thermal power from turbine heat balance data). The modified system has been discussed with the licensee and the licensee has provided a description and drawings of this modification. The modification provides increased accuracy in the range of interest to RSCS operation. The modification does not adversely affect RSCS reliability or response time. This instrument is designed for the range of possible pressure conditions in the first stage turbine for normal and abnormal operational occurrences (0 to 1500 psig). On these bases the proposed modification to the system is acceptable.

Between 50% rod density and the power level above which the RSCS is no longer necessary, the RSCS is in its group notch mode which requires that any control rod assigned to a group be kept within one notch of the other control rods in that group. Generic calculations for the RDA[3] which are directly applicable to Peach Bottom Units Nos. 2 and 3, have shown that above 20% power even multiple operator errors will not produce rod worths large enough to exceed fuel enthalpies of 280 calories/gram. The calculations are conservative in that the maximum design control rod drop speeds and Technical Specification maximum scram times were assumed, whereas for typical drop speeds and scram times a power level of ≅10% could be justified for the maximum required rated power level at which RSCS operation is required. Based on these conservatisms and others inherent in the analysis (e.g., assumption of maximum rod worth) we find the use of the ≤21% rated power level Technical Specifications for RSCS operation to be acceptable.

## III. Technical Specifications

The Commission's Office of Inspection and Enforcement has verified that the improved instrumentation has been installed on Peach Bottom Unit No. 3. Therefore, the Technical Specifications for this Unit have been issued, as proposed by the licensee. However, since the modification has not yet been installed on Unit No. 2, the current restrictions on the power level above which RSCS operation is no longer necessary has been retained at 30% with an appropriate annotation that 21% power level will apply upon completion of the modification. This staff recommended change was discussed with the licensee and he does not object. Also included in the amendments supported by this evaluation is a revised page 240e which corrects a typographical error.

IV. 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 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.

V. Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve an increase in the probability or consequences of accidents previously considered and does not involve a 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 23, 1978.

References

1. Letter from E. J. Bradley of Philadelphia Electric Company to Director of Nuclear Reactor Regulation, NRC, dated April 14, 1978.

2. Standard Review Plan, Office of Nuclear Reactor Regulation, Section 15.4.9, Spectrum of Rod Drop Accidents (BWR).

3. Stirn, R. C., et al, "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, Supplement 1, July 1972.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL

PEACH BOTTOM UNITS NOS. 2 AND 3

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 43 and 43 to Facility Operating License Nos. DPR-44 and 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 the Technical Specifications for operation of the Peach Bottom Atomic Power Station Units Nos. 2 and 3, located in York County, Pennsylvania. The amendments are effective as of the date of issuance.

These amendments revise the Technical Specifications to reduce the power level above which Rod Sequence Control System is no longer necessary. The reduction is from 30% of rated power to 21%.

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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) application for amendment dated April 14, 1978, (2) Amendments Nos. 43 and 43 to License Nos. DPR-44 and 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 single 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 23 day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors