April 7, 1

Docket Nos. 50-277
and 50-278

Mr. George A. Hunger, Jr.
Director-Licensing, MC 52A-5
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

SUBJECT: TECHNICAL SPECIFICATION CHANGE REQUESTS (TSCR) 92-06, 93-03, AND 93-04, PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 (TAC NOS. M86718 AND M86719)

The Commission has issued the enclosed Amendments Nos. 188 and 193 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3. These amendments consist of changes to the Technical Specifications in response to your application dated May 25, 1993, as supplemented on March 11, 1994.

These amendments (1) remove references to the Service Platform Hoist (TSCR 92-06), (2) correct a typographical error in the Emergency Transformer Degraded Voltage relay setpoint tolerance (TSCR 93-03), and (3) clarify the basis for recalibration of certain pressure switches (TSCR 93-04). Please inform the staff, in writing, when you have implemented these amendments.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,
/S/
Stephen Dembek, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 188 to DPR-44
2. Amendment No. 193 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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These amendments (1) remove references to the Service Platform Hoist (TSCR 92-06), (2) correct a typographical error in the Emergency Transformer Degraded Voltage relay setpoint tolerance (TSCR 93-03), and (3) clarify the basis for recalibration of certain pressure switches (TSCR 93-04). Please inform the staff, in writing, when you have implemented these amendments.

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1. Amendment No. 188 to DPR-44
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See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 188
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, et. al. (the licensee) dated May 25, 1993, as supplemented on March 11, 1994, 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 or 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. 188 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 188

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 areas are indicated by marginal lines.

Remove	Insert
39	39
49	49
50	50
71b	71b
226	226
229	229

NOTES FOR TABLE 3.1.1

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable sensor channels for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place mode switch in the start up position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30% rated.

2. Permissible to bypass, in refuel and shutdown positions of the reactor mode switch.

3. Deleted.

4. Bypassed when reactor thermal power is less than 30% of rated as indicated by turbine first stage pressure.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212 degrees F, only the following trip functions need to be operable.

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge instrument volume high level

8. Not required to be operable when primary containment integrity is not required.

9. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

3.1 BASES (Cont'd)

the amount of water which must be accomodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accomodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accomodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accomodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions (reference paragraph 7.5.4 FSAR). Thus, the IRM and APRM are required in the "Refuel" and "Start/Hot Standby" modes. In the power range the APRM system provides required protection (reference paragraph 7.5.7 FSAR). Thus the IRM System is not required in the "Run" mode. The APRM's cover only the power range. The IRMs and APRMs provide adequate coverage in the start-up and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for Startup and Run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions indicated in Table 3.1.1 operable in the Refuel mode assures that shifting to the Refuel mode during reactor power operation does not diminish the protection provided by the reactor protection system.

The turbine condenser low vacuum scram is only required during power operation and must be bypassed to start up the unit. The main condenser low vacuum trip is bypassed except in the run position of the mode switch.

Turbine stop valve closure occurs at 10% of valve closure. Below 30% of rated reactor thermal power the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

3.1 BASES (Cont'd.)

Turbine control valves fast closure initiates a scram based on pressure switches sensing Electro-Hydraulic Control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative (500<P<850 psig) to the normal EHC oil pressure of 1600 psig gauge that, based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure. This scram signal is also bypassed when reactor thermal power is less than 30% of rated as indicated by turbine first stage pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the Startup and Refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

TABLE 3.2.B (CONTINUED)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2 per 4 kV Bus	Emergency Transformer Degraded voltage (Inverse time - voltage). (CV-6)	87% (±5%) of Rated Voltage. Tests at 2940 volts in 30 seconds (±10%)		1. Trips emergency transformer feed to 4kV emergency bus. 2. Fast transfer permissive.

LIMITING CONDITIONS FOR OPERATION

3.10.A Refueling Interlocks

3. The fuel grapple hoist load switch shall be set at ≤ 1000 lbs.

4. If the frame-mounted auxiliary hoist or the monorail-mounted auxiliary hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at ≤ 400 lbs.

5. A maximum of two nonadjacent control rods may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance, provided the following conditions are satisfied:

   a. The reactor mode switch shall be locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be operable.

   b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all

SURVEILLANCE REQUIREMENTS

4.10.A.2 (Cont'd)

fully withdrawn and all other operable rods fully inserted. Alternatively if the remaining control rods are fully inserted and have their directional control valves electrically dis-armed, it is sufficient to demonstrate that the core is subcritical with a margin of at least 0.25% Δ k at any time during the main-tenance. A control rod on which maintenance is being performed shall be considered inoperable.

3.10 BASES

A. Refueling Interlocks

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. The interlocks prevent an inadvertent criticality during refueling operations when the reactivity potential of the core is being altered.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment.

The refueling interlocks include circuitry which senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated which prevent the movement of the refueling equipment or withdrawal of control rods (rod block).

Circuitry is provided which senses the following conditions:

1. All rods inserted.

2. Refueling platform positioned near or over the core.

3. Refueling platform hoists are fuel-loaded (fuel grapple, frame mounted hoist, monorail mounted hoist).

4. Fuel grapple not full up.

5. Deleted.

6. One rod withdrawn.

When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position, only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor

UNITED STATES
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WASHINGTON, D.C. 20555-0001

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. 193
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, et. al. (the licensee) dated May 25, 1993, as supplemented on March 11, 1994, 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 or 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.193 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 193

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 areas are indicated by marginal lines.

Remove	Insert
39	39
49	49
50	50
71b	71b
226	226
229	229

NOTES FOR TABLE 3.1.1

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable sensor channels for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place mode switch in the start up position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30% rated.

2. Permissible to bypass, in refuel and shutdown positions of the reactor mode switch.

3. Deleted.

4. Bypassed when reactor thermal power is less than 30% of rated as indicated by turbine first stage pressure.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212 degrees F, only the following trip functions need to be operable.

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge instrument volume high level

8. Not required to be operable when primary containment integrity is not required.

9. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

3.1 BASES (Cont'd)

the amount of water which must be accomodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accomodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accomodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accomodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions (reference paragraph 7.5.4 FSAR). Thus, the IRM and APRM are required in the "Refuel" and "Start/Hot Standby" modes. In the power range the APRM system provides required protection (reference paragraph 7.5.7 FSAR). Thus the IRM System is not required in the "Run" mode. The APRM's cover only the power range. The IRMs and APRMs provide adequate coverage in the start-up and intermediate range.

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The turbine condenser low vacuum scram is only required during power operation and must be bypassed to start up the unit. The main condenser low vacuum trip is bypassed except in the run position of the mode switch.

Turbine stop valve closure occurs at 10% of valve closure. Below 30% of rated reactor thermal power the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

 Amendment No. ~~121~~, 193

3.1 BASES (Cont'd.)

Turbine control valves fast closure initiates a scram based on pressure switches sensing Electro-Hydraulic Control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative (500<P<850 psig) to the normal EHC oil pressure of 1600 psig gauge that, based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure. This scram signal is also bypassed when reactor thermal power is less than 30% of rated as indicated by turbine first stage pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the Startup and Refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

TABLE 3.2.B (CONTINUED)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2 per 4 kV Bus	Emergency Transformer Degraded voltage (Inverse time - voltage). (CV-6)	87% (±5%) of Rated Voltage. Tests at 2940 volts in 30 seconds (±10%)		1. Trips emergency transformer feed to 4kV emergency bus. 2. Fast transfer permissive.

LIMITING CONDITIONS FOR OPERATION

3.10.A Refueling Interlocks

3. The fuel grapple hoist load switch shall be set at ≤ 1000 lbs.

4. If the frame-mounted auxiliary hoist or the monorail-mounted auxiliary hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at ≤ 400 lbs.

5. A maximum of two nonadjacent control rods may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance, provided the following conditions are satisfied:

   a. The reactor mode switch shall be locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be operable.

   b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all

SURVEILLANCE REQUIREMENTS

4.10.A.2 (Cont'd)

fully withdrawn and all other operable rods fully inserted. Alternatively if the remaining control rods are fully inserted and have their directional control valves electrically disarmed, it is sufficient to demonstrate that the core is subcritical with a margin of at least 0.25% Δ k at any time during the maintenance. A control rod on which maintenance is being performed shall be considered inoperable.

3.10 BASES

A. Refueling Interlocks

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. The interlocks prevent an inadvertent criticality during refueling operations when the reactivity potential of the core is being altered.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during the refueling operations by restricting the movement of control rods and the operation of refueling equipment.

The refueling interlocks include circuitry which senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated which prevent the movement of the refueling equipment or withdrawal of control rods (rod block).

Circuitry is provided which senses the following conditions:

1. All rods inserted.
2. Refueling platform positioned near or over the core.
3. Refueling platform hoists are fuel-loaded (fuel grapple, frame mounted hoist, monorail mounted hoist).
4. Fuel grapple not full up.
5. Deleted.
6. One rod withdrawn.

When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position, only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 188 AND 193 TO FACILITY OPERATING

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, UNIT NOS. 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By letter dated May 25, 1993, as supplemented on March 11, 1994, the Philadelphia Electric Company (the licensee) submitted a request for changes to the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, Technical Specifications (TS). The requested changes include the administrative revisions in Technical Specification Change Requests (TSCR) 92-06, 93-03 and 93-04. These TSCRs request (1) removal of references to the Service Platform Hoist (TSCR 92-06), (2) correction of a typographical error concerning the Emergency Transformer Degraded Voltage relay setpoint tolerance (TSCR 93-03), and (3) clarification of recalibration basis of certain pressure switches (i.e., basis should be reactor thermal power instead of turbine first stage pressure (TSCR 93-04)). The March 11, 1994 letter corrected typgrophical errors in the licensee's May 25, 1993 submittal and did not change the original no significant hazards consideration.

2.0 EVALUATION

The following TS changes were proposed by the licensee:

TSCR 92-06

(1) Delete the wording "or the service platform hoist", and add the word "or" in Limiting Condition for Operation (LCO) 3.10.A.4, on page 226;

(2) Delete item #5 on 3.10 BASES, A. Refueling Interlocks, on page 229;

(3) Delete Surveillance Requirement 4.10.A.3, on page 226; (Unit 3 only)

(4) Delete asterisk and corresponding footnote in second paragraph of 3.10 BASES, A. Refueling Interlocks, on page 229. (Unit 3 only)

TSCR 93-03

(1) Add a minus sign to the setpoint tolerance of the Emergency Transformer Degraded Voltage Relay.

TSCR 93-04

(1) Revise note 4 on Notes for Table 3.1.1, page 39, to read:

"Bypassed when reactor thermal power is less than 30% of rated as indicated by turbine first stage pressure."

(2) Revise wording of last paragraph in BASES Section 3.1, page 49, to read:

"Turbine stop valve closure occurs at 10% of valve closure. Below 30% of rated reactor thermal power the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor."

(3) Revise the last sentence of the first paragraph of BASES Section 3.1 (cont'd), page 50, to read:

"This scram signal is also bypassed when reactor thermal power is less than 30% of rated as indicated by turbine first stage pressure."

The changes are administrative in nature and serve to improve record keeping based on:

TSCR 92-06

(1), (2) The Service Platform Hoist and its associated load sensing switches were removed in 1985. In addition, the refueling interlock circuit which prevents the operation of the Service Platform when its hoist is loaded has been altered by modification 5221. Since the Service Platform Hoist no longer exists, the applicable TSs need to be revised in accordance with the present plant configuration.

(3), (4) The Surveillance Requirement 4.10.A.3, on page 226, and the asterisk and corresponding footnote in second paragraph of 3.10 BASES, A. Refueling Interlocks, on page 229, Unit 3, may be deleted since these TS provisions applied to Unit 3 prior to tensioning the reactor vessel head bolts for the cycle 8 refueling outage. Unit 3 is currently operating in cycle 10. Therefore, these changes will remove obsolete TS provisions.

TSCR 93-03

(1) The absence of the minus sign to the setpoint tolerance of the Emergency Transformer Degraded Voltage Relay is a typographical error which has

existed since the issuance of amendment numbers 97 (Unit 2) and 99 (Unit 3) to the TS. The licensee submitted this TSCR to correct the typographical error.

TSCR 93-04

(1), (2), (3) This is a basis change for establishing the scram signal bypass setpoint for Turbine Control Valve Fast Closure (TCVFC) and Turbine Stop Valve Closure (TSVC). The licensee is responding to General Electric Company's (GE) Service Information Letter (SIL) 423 for this change. GE SIL 423 includes a recommendation for licensees to review their TS to ensure that the correct power bases are clearly stated. Upon reviewing the applicable sections, the licensee determined that the wording needed to be revised to eliminate confusion when interpreting the TS.

Based on the above discussion, the staff finds TSCR 92-06, 93-03 and 93-04 are administrative changes that enhance the clarity of the TS. Therefore, the proposed changes are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 39059). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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