March 1, 1983

DMB 016

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Dockets Nos. 50-277
and 50-278

Mr. Edward G. Bauer, Jr.
Vice President and General Counsel
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Dear Mr. Bauer:

Re: Peach Bottom Atomic Power Station, Units 2 and 3

The Commission has issued the enclosed Amendments Nos. 88 and 88 to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 14, 1980, as supplemented October 7, 1981.

The amendments revise the TSs to provide limiting conditions for operation and surveillance requirements for scram discharge volume (SDV) vent and drain valves and reactor protection system and control rod block SDV limit switches.

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

Sincerely,

Original signed by

Gerald Gears, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 88 to DPR-44
2. Amendment No. 88 to DPR-56
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

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ORB-2

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD	Ken Eccleston	
SURNAME	RIngram	GGears:cf	JStolz	GLainas	Cutchin	KE	
DATE	2/15/82	2/9/83	2/16/83	2/17/83	2/25/83	2/17/83	

NRC FORM 318 (10-80) NRCM 0240 — OFFICIAL RECORD COPY — USGPO: 1981—335-960

Philadelphia Electric Company

cc w/enclosure(s):

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Philadelphia Electric Company
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Delta, Pennsylvania 17314

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Board of Supervisors
Peach Bottom Township
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Delta, Pennsylvania 17314

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Governor's Office of State Planning
and Development
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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. 88
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 October 14, 1980, as supplemented October 7, 1981, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 88 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 1, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 88

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
42	42
73	73
—	74a (new page)
83	83*
92	92
100	100
112	112

*Overleaf page provided for document completeness.

TABLE 4.1.1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
High Water Level In Scram Discharge Tank	A	Trip Channel and Alarm	Every 1 month.
Turbine Condenser Low Vacuum (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1).
Main Steam Line High Radiation	B1	Trip Channel and Alarm (4)	Once/week.
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Every 1 month (1).
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Every 1 month.
Turbine First Stage Pressure Permissive	A	Trip Channel and Alarm	Every 3 months (1).
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Every 1 month (1).
Reactor Pressure Permissive (6)	B2	Trip Channel and Alarm (4)	Every 3 months.

Amendment No. ~~25~~, ~~34~~, ~~38~~, ~~42~~, ~~46~~, ~~70~~, 78, 88

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq (0.66W+42-0.66\Delta W) \times \frac{FRP}{MFLPD}$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Startup Mode)	$\leq$12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq (0.66W+41-0.66\Delta W) \times \frac{FRP}{MFLPD}$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (3)	$\geq$2.5 indicated on scale	8 Inst. Channels	(1)
3	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$108 indicated on scale	8 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Volume High Level	$\leq$ .25 gallons	1 Inst. Channel	(9)

Unit 2

NOTES FOR TABLE 3.2.C (Cont.)

9. If the number of operable channels is less than required by the minimum operable per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn.

TABLE 4.2.C

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
1) APRM - Downscale	(1) (3)	Once/3 months	Once/day
2) APRM - Upscale	(1) (3)	Once/3 months	Once/day
3) IRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
4) IRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
5) RBM - Upscale	(1) (3)	Once/6 months	Once/day
6) RBM - Downscale	(1) (3)	Once/6 months	Once/day
7) SRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
8) SRM - Detector Not in Startup Position	(2) (3)	Startup or Control Shutdown	(2)
9) IRM - Detector Not in Startup Position	(2) (3)	Startup or Control Shutdown	(2)
10) Scram Discharge Volume - High Level	Quarterly	Once/Operating Cycle	NA

Logic System Functional Test (4) (6)	Frequency
(1) System Logic Check	Once/6 months

TABLE 4.2.D

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS

Instrument Channels	Instrument Functional Test	Calibration	Instrument Check (2)
1) Refuel Area Exhaust Monitors - Upscale	(1)	Once/3 months	Once/day
2) Reactor Building Area Exhaust Monitors - Upscale	(1)	Once/3 months	Once/day
3) Off-Gas Radiation Monitors	(1)	Once/3 months	Once/day

Logic System Functional Test (4) (6)	Frequency
1) Reactor Building Isolation	Once/6 months
2) Standby Gas Treatment System Actuation	Once/6 months
3) Steam Jet Air Ejector Off-Gas Line Isolation	Once/6 months

3.2 BASES (Cont'd)

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequences. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit.

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in the local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in the control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

High water level in the scram discharge volume may be indicative of excessive scram valve leakage, or plugging or closing of the discharge volume drain valve, and could jeopardize the ability of all rods to fully insert on a scram signal.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function (ADS) is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale

LIMITING CONDITIONS FOR OPERATION

3.3.A Reactivity Limitations (Cont'd)

failure is not due to a failed control rod drive mechanism collet housing.

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.

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.

SURVEILLANCE REQUIREMENTS

4.3.A Reactivity Limitations (Cont'd)

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 scram discharge volume drain and vent valves shall be verified open at least once per month. These valves may be closed intermittently for testing.

c. At least once every 3 months verify that the scram discharge volume drain and vent valves closed within 15 seconds after receipt of a closure signal, and reopen upon reset of the closure signal.

d. 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.

3.3 & 4.3 BASES (Cont'd)

identified as the resistance to drive motion by an internal control rod drive filter. The filter had been loaded by foreign material, probably accelerated by construction debris. The sudden changes in drive scram performance which were observed at that plant were due to stepwise release into reactor coolant of particulate matter as the reactor and subsystem were subsequently started up. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Dresden 2 has currently 27 "B" type drives. Approximately 4718 drive tests have been recorded to date. Data documenting the successful performance of the modified drive has been submitted to the NRC with a letter from Commonwealth Edison Company to the Commission dated November 6, 1972 with the subject of the letter being Proposed Changes to Quad-Cities Power Station Operating License, including Appendices A and B, DPR 29 and 30, AEC Dkts 50-254 and 50-265.

Although the cause and cure of the dirt problem were known at the time of the writing of the Dresden 3 Tech Specs, the progressive surveillance requirement was incorporated into the technical specification to ostensibly detect any other unforeseen drive problems. The possibility of this being a temporary requirement may be inferred from the provision for review of all surveillance requirements after the first operating cycle.

Operability of the scram discharge volume vent and drain valves is necessary for maintaining a reservoir to contain the water exhausted from all control rod drives during a scram.

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. 88
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 October 14, 1980, as supplemented October 7, 1981, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 88 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 1, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 88

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
42	42
73	73
—	74a (new page)
83	83*
92	92
100	100
112	112

*Overleaf page provided for document completeness.

TABLE 4.1.1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
High Water Level In Scram Discharge Tank	A	Trip Channel and Alarm	Every 1 month.
Turbine Condenser Low Vacuum (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1).
Main Steam Line High Radiation	B1	Trip Channel and Alarm (4)	Once/week.
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Every 1 month (1).
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Every 1 month.
Turbine First Stage Pressure Permissive	A	Trip Channel and Alarm	Every 3 months (1).
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Every 1 month (1).
*Reactor Pressure Permissive (6)	B2	Trip Channel and Alarm (4)	Every 3 months.
**Reactor Pressure Permissive	A	Trip Channel and Alarm	Every 3 months.

* Deleted when modifications authorized by Amendment No. 67 are completed.

** Effective when modifications authorized by Amendment No. 67 are completed.

Amendment No. 67, 88

Amendment No. 33, 62, 77, 88

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq (0.66W+42-0.66\Delta W) \times \frac{FRP}{MFLPD}$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Startup Mode)	$\leq$12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq (0.66W+41-0.66\Delta W) \times \frac{FRP}{MFLPD}$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (3)	$\geq$2.5 indicated on scale	8 Inst. Channels	(1)
3	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$108 indicated on scale	8 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Volume High Level	$\leq$.25 gallons	1 Inst. Channel	(9)

NOTES FOR TABLE 3.2.C (Cont.)

9. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, "Startup" mode and "Refuel" mode if more than one control rod is withdrawn.

TABLE 4.2.C

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
1) APRM - Downscale	(1) (3)	Once/3 months	Once/day
2) APRM - Upscale	(1) (3)	Once/3 months	Once/day
3) IRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
4) IRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
5) RBM - Upscale	(1) (3)	Once/6 months	Once/day
6) RBM - Downscale	(1) (3)	Once/6 months	Once/day
7) SRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
8) SRM - Detector Not in Startup Position	(2) (3)	Startup or Control Shutdown	(2)
9) IRM - Detector Not in Startup Position	(2) (3)	Startup or Control Shutdown	(2)
10) Scram Discharge Volume - High Level	Quarterly	Once/Operating Cycle	NA

Logic System Functional Test (4) (6)	Frequency
(1) System Logic Check	Once/6 months

TABLE 4.2.D

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS

Instrument Channels	Instrument Functional Test	Calibration	Instrument Check (2)
1) Refuel Area Exhaust Monitors - Upscale	(1)	Once/3 months	Once/day
2) Reactor Building Area Exhaust Monitors - Upscale	(1)	Once/3 months	Once/day
3) Off-Gas Radiation Monitors	(1)	Once/3 months	Once/day

Logic System Functional Test (4) (6)	Frequency
1) Reactor Building Isolation	Once/6 months
2) Standby Gas Treatment System Actuation	Once/6 months
3) Steam Jet Air Ejector Off-Gas Line Isolation	Once/6 months

3.2 BASES (Cont'd)

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequences. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit.

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in the local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in the control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

High water level in the scram discharge volume may be indicative of excessive scram valve leakage, or plugging or closing of the discharge volume drain valve, and could jeopardize the ability of all rods to fully insert on a scram signal.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function (ADS) is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale

LIMITING CONDITIONS FOR OPERATION

3.3.A Reactivity Limitations (Cont'd)

failure is not due to a failed control rod drive mechanism collet housing.

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.

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.

SURVEILLANCE REQUIREMENTS

4.3.A Reactivity Limitations (Cont'd)

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 scram discharge volume drain and vent valves shall be verified open at least once per month. These valves may be closed intermittently for testing.

c. At least once every 3 months verify that the scram discharge volume drain and vent valves closed within 15 seconds after receipt of a closure signal, and reopen upon reset of the closure signal.

d. 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.

3.3 & 4.3 BASES (Cont'd)

identified as the resistance to drive motion by an internal control rod drive filter. The filter had been loaded by foreign material, probably accelerated by construction debris. The sudden changes in drive scram performance which were observed at that plant were due to stepwise release into reactor coolant of particulate matter as the reactor and subsystem were subsequently started up. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Dresden 2 has currently 27 "B" type drives. Approximately 4718 drive tests have been recorded to date. Data documenting the successful performance of the modified drive has been submitted to the NRC with a letter from Commonwealth Edison Company to the Commission dated November 6, 1972 with the subject of the letter being Proposed Changes to Quad-Cities Power Station Operating License, including Appendices A and B, DPR 29 and 30, AEC Dkts 50-254 and 50-265.

Although the cause and cure of the dirt problem were known at the time of the writing of the Dresden 3 Tech Specs, the progressive surveillance requirement was incorporated into the technical specification to ostensibly detect any other unforeseen drive problems. The possibility of this being a temporary requirement may be inferred from the provision for review of all surveillance requirements after the first operating cycle.

Operability of the scram discharge volume vent and drain valves is necessary for maintaining a reservoir to contain the water exhausted from all control rod drives during a scram.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENTS NOS. 88 AND 88 TO FACILITY OPERATING LICENSES 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

Introduction

By letter dated October 14, 1980, as supplemented October 7, 1981, the Philadelphia Electric Company (the licensee) proposed an amendment to the Technical Specifications (TSs) appended to Facility Operating Licenses Nos. DPR-44 and DPR-56. The proposed amendment would provide surveillance requirements for scram discharge volume (SDV) vent and drain valves and limiting conditions for operation /surveillance requirements for reactor protection system (RPS) and control block SDV limit switches.

Background

As a result of events involving common cause failures of SDV limit switches and SDV drain valve operability, the NRC staff issued IE Bulletin 80-14 on June 12, 1980. In addition, the NRC staff sent a letter dated July 7, 1980, to all operating BWR licensees requesting that they propose TS changes to provide surveillance requirements for SDV vent and drain valves and limiting conditions for operation/ surveillance requirements on SDV limit switches. Model TSs were enclosed with this letter to provide guidance to licensees for preparation of the requested submittals.

Evaluation

The enclosed Technical Evaluation Report (TER-C5506-68/72) was prepared for us by Franklin Research Center (FRC) as part of our technical assistance contract program. Their report provides their technical evaluation of the compliance of the licensee's submittal with NRC provided criteria.

FRC has concluded that the licensee's response does not meet the explicit requirements of paragraph 3.3-6 and Table 3.3.6-1 of the NRC staff's Model TSs. However, the FRC report concludes that technical bases are defined on p. 50 of the staff's "Generic Safety Evaluation Report BWR Scram Discharge System," December 1, 1980 for this departure from the explicit requirements of the Model TSs.

A summary of the evaluation for those areas where the licensee's response represents a departure from the explicit requirements of the Model TSs is provided below:

> The alarm and rod block instrumentation, consisting of one operable instrument channel with one limit switch for control rod withdrawal block as specified on the revised TS page 73, is acceptable because the licensee's long term modification of the scram discharge system provides for an adequate and acceptable hydraulic coupling between scram discharge headers and instrumented volume.
>
> Because the "Scram Discharge Volume Scram Trips" cannot be bypassed at the Peach Bottom Atomic Power Station, Units 2 and 3, while the reactor is in operational conditions of startup and run, and interlocks are provided which prevent the withdrawal of more than one control rod with the mode switch in the refuel position, the Model TSs (paragraph 3.3.6, Table 3.3.6-1, paragraph 4.3.6 and Table 4.3.6-1) are, therefore, not applicable for "Trip Function 5b, SDV Scram Trip Bypassed."

We conclude that these technical bases justify a deviation from the explicit requirements of the Model TSs.

FRC has concluded that the licensee's proposed TS revisions meet our criteria without the need for further revision.

Summary

Based upon our review of the contractor's report and discussions with the reviewer, we conclude that the licensee's proposed TSs satisfy our requirements for surveillance of SDV vent and drain valves and for limiting conditions for operation/surveillance requirements for SDV limit switches. Consequently, we find the licensee's proposed TSs acceptable for Peach Bottom Units 2 and 3.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve 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 these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 1, 1983

Enclosure:
TER

The following NRC personnel have contributed to this Safety Evaluation:
Gerry Gears, Ken Eccleston.

TECHNICAL EVALUATION REPORT

BWR SCRAM DISCHARGE VOLUME LONG-TERM MODIFICATIONS

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

NRC DOCKET NO. 50-277, 50-278	FRC PROJECT C5506
NRC TAC NO. 42217, 42216	FRC ASSIGNMENT 2
NRC CONTRACT NO. NRC-03-81-130	FRC TASKS 68, 72

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December 22, 1981

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CONTENTS

SUMMARY

This technical evaluation report reviews and evaluates Phase 1 proposed changes in the Peach Bottom Atomic Power Station Units 2 and 3 Technical Specifications for scram discharge volume (SDV) long-term modifications regarding surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for reactor protection system and control rod withdrawal block SDV limit switches. Conclusions were based on the degree of compliance of the Licensee's original submittal and response to a RFI with criteria from the Nuclear Regulatory Commission (NRC) staff's Model Technical Specifications.

Proposed revisions of pages 42, 73, 74, 83, 100, and 112, and unrevised pages 38, 39, 43, 44, 46, and 111 of the Peach Bottom Atomic Power Station Units 2 and 3 Technical Specifications fully meet the surveillance requirements of the NRC staff's Model Technical Specifications. Table 5-1 on pages 25 and 26 summarizes the evaluation results.

1. INTRODUCTION

1.1 PURPOSE OF THE TECHNICAL EVALUATION

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of the Peach Bottom Atomic Power Station Units 2 and 3 boiling water reactor (BWR) in regard to "BWR Scram Discharge Volume Long Term Modification," specifically:

- o surveillance requirements for scram discharge volume (SDV) vent and drain valves
- o limiting condition for operation (LCO)/surveillance requirements for the reactor protection system
- o LCO/surveillance requirements for the control rod withdrawal block SDV limit switches.

The evaluation used criteria proposed by the NRC staff in Model Technical Specifications (see Appendix A of this report). This effort is directed toward the NRC objective of increasing the reliability of installed BWR scram discharge volume systems, the need for which was made apparent by events described below.

1.2 GENERIC ISSUE BACKGROUND

On June 13, 1979, while the reactor at Hatch Unit 1 was in the refuel mode, two SDV high level switches had been modified, tested, and found inoperable. The remaining switches were operable. Inspection of each inoperable level switch revealed a bent float rod binding against the side of the float chamber.

On October 19, 1979, Brunswick Unit 1 reported that water hammer due to slow closure of the SDV drain valve during a reactor scram damaged several pipe supports on the SDV drain line. Drain valve closure time was approximately 5 minutes because of a faulty solenoid controlling the air supply to the valve. After repair, to avoid probable damage from a scram, the unit was started with the SDV vent and drain valves closed except for periodic draining. During this mode of operation, the reactor scrammed due to a high water level in the

SDV system without prior actuation of either the high level alarm or rod block switch. Inspection revealed that the float ball on the rod block switch was bent, making the switches inoperable. The water hammer was reported to be the cause of these level switch failures.

As a result of these events involving common-cause failures of SDV limit switches and SDV drain valve operability, the NRC issued IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capability," on June 12, 1980 [1]. In addition, to strengthen the provisions of this bulletin and to ensure that the scram system would continue to work during reactor operation, the NRC sent a letter dated July 7, 1980 [2] to all operating BWR licensees requesting that they propose Technical Specifications changes to provide surveillance requirements for reactor protection system and control rod block SDV limit switches. The letter also contained the NRC staff's Model Technical Specifications to be used as a guide by licensees in preparing their submittals.

Meanwhile, during a routine shutdown of the Browns Ferry Unit 3 reactor on June 28, 1980, 76 of 185 control rods failed to insert fully. Full insertion required two additional manual scrams and an automatic scram for a total elapsed time of approximately 15 minutes between the first scram initiation and the complete insertion of all the rods. On July 3, 1980, in response to both this event and the previous events at Hatch Unit 1 and Brunswick Unit 1, the NRC issued (in addition to the earlier IE Bulletin 80-14) IE Bulletin 80-17 [3] followed by five supplements [4-8]. These initiated short-term and long-term programs described in "Generic Safety Evaluation Report BWR Scram Discharge System," NRC Staff, December 1, 1980 [9] and "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17 (Continuous Monitoring Systems)" [10].

Analysis and evaluation of the Browns Ferry Unit 3 and other SDV system events convinced the NRC staff that SDV systems in all BWRs should be modified to assure long-term SDV reliability. Improvements were needed in three major areas: SDV-IV hydraulic coupling, level instrumentation, and system isolation. To achieve these objectives, an Office of Nuclear Reactor Regulation (NRR) task force and a subgroup of the BWR Owners Group developed Revised Scram Discharge

System Design and Safety Criteria for use in establishing acceptable SDV systems modifications [9]. Also, an NRC letter dated October 1, 1980 requested all operating BWR licensees to reevaluate installed SDV systems and modify them as necessary to comply with the revised criteria.

In Reference 9, the SDV-IV hydraulic coupling at the Big Rock Point, Brunswick 1 & 2, Duane Arnold, and Hatch 1 & 2 BWRs was judged acceptable. The remaining BWRs will require modification to meet the revised SDV-IV hydraulic coupling criteria, and all operating BWRs may require modification to meet the revised instrumentation and isolation criteria. The changes in Technical Specifications associated with this effort will be carried out in two phases:

Phase 1 - Improvements in surveillance for vent and drain valves and instrument volume level switches.

Phase 2 - Technical Specifications improvements required as a result of long-term modifications made to comply with revised design and performance criteria.

This TER is a review and evaluation of Technical Specifications changes proposed for Phase 1.

1.3 PLANT-SPECIFIC BACKGROUND

The July 7, 1980 NRC letter [2] not only requested all BWR licensees to amend their facilities' Technical Specifications with respect to control rod drive SDV capability, but enclosed the NRC staff's proposed Model Technical Specifications (see Appendix A of this TER) as a guide for the licensees in preparing the requested submittals and as a source of criteria for an FRC technical evaluation of the submittals. In this TER, FRC has reviewed and evaluated Technical Specifications changes for the Peach Bottom Atomic Power Station Units 2 and 3 proposed by the Licensee, Philadephia Electric Company (PECO), in an October 14, 1980 letter with the initial submittal (see Appendix B) and in an October 7, 1981 letter with the response to a RFI (see Appendix C) in regard to "BWR Scram Discharge Volume (SDV) Long-Term Modifications" and, specifically, the surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for the reactor protection system and control rod withdrawal block SDV limit switches. FRC assessed the adequacy with which the PECO information documented compliance of the proposed Technical Specifications changes with the NRC staff's Model Technical Specifications.

2. REVIEW CRITERIA

The criteria established by the NRC staff's Model Technical Specifications involving surveillance requirements of the main SDV components and instrumentation cover three areas of concern:

- o surveillance requirements for SDV vent and drain valves
- o LCO/surveillance requirements for reactor protection system SDV limit switches
- o LCO/surveillance requirements for control rod block SDV limit switches.

2.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

The surveillance criteria of the NRC staff's Model Technical Specifications for SDV drain valves are:

"4.1.3.1.1 - The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days, and
b. Cycling each valve at least one complete cycle of full travel at least once per 92 days.

*These valves may be closed intermittently for testing under administrative controls."

The Model Technical Specifications require testing the drain and vent valves, checking at least once every 31 days that each valve is fully open during normal operation, and cycling each valve at least one complete cycle of full travel under administrative controls at least once per 92 days.

Full opening of each valve during normal operation indicates that there is no degradation in the control air system and its components that control the air pressure to the pneumatic actuators of the drain and vent valves. Cycling each valve checks whether the valve opens fully and whether its movement is smooth, jerky, or oscillatory.

During normal operation, the drain and vent valves stay in the open position for very long periods. A silt of particulates such as metal chips and flakes, various fibers, lint, sand, and weld slag from the water or air may accumulate at moving parts of the valves and temporarily "freeze" them. A strong breakout force may be needed to overcome this temporary freeze, producing a violent jerk which may induce a severe water hammer if it occurs during a scram or a scram resetting. Periodic cycling of the drain and vent valves is the best method to clear the effects of particulate silting, thus promoting smooth opening and closing and more reliable valve operation. Also, in case of improper valve operation, cycling can indicate whether excessive pressure transients may be generated during and after a reactor scram which might damage the SDV piping system and cause a loss of system integrity or function.

2.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

The paragraphs of the NRC staff's Model Technical Specifications pertinent to LCO/surveillance requirements for reactor protection system SDV limit switches are:

"3.3.1 - As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

Table 3.3.1-1. Reactor Protection System Instrumentation

Functional Unit	Applicable Operational Conditions	Minimum Operable Channels Per Trip System (a)	Action
8. Scram Discharge Volume Water Level-High	1,2,5 (h)	2	4

Table 3.3.1-2. Reactor Protection System Response Times

Functional Unit	Response Time (Seconds)
8. Scram Discharge Volume Water Level-High	NA"

"4.3.1.1 - Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

Table 4.3.1.1-1. Reactor Protection System Instrumentation Surveillance Requirements

Functional Unit	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
8. Scram Discharge Volume Water Level-High	NA	M	R	1,2,5

Notation (a) A channel may be placed in an inoperable status up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2

Action 4: In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2."

Paragraph 3.3.1 and Table 3.3.1-1 of the Model Technical Specifications require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, a total of 4 operable channels containing 4 limit switches per 2 trip systems for the reactor protection system which automatically initiates a scram. The technical objective of these requirements is to provide 1-out-of-2-taken-twice logic for the reactor protection system. The response time of the reactor protection system for the functional unit of SDV water level-high should be measured and kept available (it is not given in Table 3.3.1-2).

Paragraph 4.3.1.1 and Table 4.3.1.1-1 give reactor protection system instrumentation surveillance requirements for the functional unit of SDV water level-high. Each reactor protection system instrumentation channel containing a limit switch should be shown to be operable by the Channel Functional Test monthly and Channel Calibration at each refueling outage.

2.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

The NRC staff's Model Technical Specifications specify the following LCO/surveillance requirements for control rod withdrawal block SDV limit switches:

"3.3.6 - The control rod withdrawal block instrumentation channel shown in Table 3.3.6-1 shall be OPERABLE with trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

Table 3.3.6-1. Control Rod Withdrawal Block Instrumentation

Trip Function	Minimum Operable Channels Per Trip Function	Applicable Operational Conditions	Action
5. Scram Discharge Volume			
a. Water level-high	2	1, 2, 5**	62
b. Scram trip bypassed	1	(1, 2, 5**)	62

ACTION 62: With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Table 3.3.6-2. Control Rod Withdrawal Block Instrumentation Setpoints

Trip Function	Trip Setpoint	Allowable Value
5. Scram Discharge Volume		
a. Water level-high	To be specified	NA
b. Scram trip bypassed	NA	NA*

"4.3.6. Each of the above control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

Table 4.3.6-1. Control Rod Withdrawal Block Instrumentation Surveillance Requirements

Trip Function	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
5. Scram Discharge Volume				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2."

Paragraph 3.3.6 and Table 3.3.6-1 of the Model Technical Specifications require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high and 1 operable channel containing 1 limit switch for SDV scram trip bypassed. The technical objective of these requirements is to have at least one channel containing one limit switch available to monitor the SDV water level when the other channel with a limit switch is being tested or undergoing maintenance. The trip setpoint for control rod withdrawal block instrumentation monitoring

SDV water level-high should be specified as indicated in Table 3.3.6-2. The trip function prevents further withdrawal of any control rod when the control rod block SDV limit switches indicate water level-high.

Paragraph 4.3.6 and Table 4.3.6-1 require that each control rod withdrawal block instrumentation channel containing a limit switch be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

The Surveillance Criteria of the BWR Owners Subgroup given in Appendix A, "Long-Term Evaluation of Scram Discharge System," of "Generic Safety Evaluation Report BWR Scram Discharge System," written by the NRC staff and issued on December 1, 1980, are:

1. Vent and drain valves shall be periodically tested.
2. Verifying and level detection instrumentation shall be periodically tested in place.
3. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control rod density.

Analysis of the above criteria indicates that the NRC staff's Model Technical Specifications requirements, the acceptance criteria for the present TER, fully cover the BWR Owners Subgroup Surveillance Criteria 1 and 2 and partially cover Criterion 3.

3. METHOD OF EVALUATION

The PECO submittal for the Peach Bottom Atomic Power Station Units 2 and 3 was evaluated in two stages, initial and final.

During the initial evaluation, only the NRC staff's Model Technical Specifications requirements were used to determine if:

- o the Licensee's submittal was responsive to the July 7, 1980 NRC request for proposed Technical Specifications changes involving the surveillance requirements of the SDV vent and drain valves, LCO/surveillance requirements for reactor protection system SDV limit switches, and LCO/surveillance requirements for control rod block SDV limit switches
- o the submitted information was sufficient to permit a detailed technical evaluation.

During the final evaluation, in addition to the NRC staff's Model Technical Specifications requirements, background material in References 1 through 10, pertinent sections of "Philadelphia Electric Company Peach Bottom Atomic Power Station Units 2 and 3 Final Safety Analysis Report," and Peach Bottom Technical Specifications were studied to determine the technical bases for the design of SDV main components and instrumentation. Subsequently, the Licensee's response was compared directly to the requirements of the NRC staff's Model Technical Specifications. The findings of the final evaluation are presented in Section 4 of this report.

The initial evaluation concluded that the Licensee's submittal was responsive to the NRC request of July 7, 1980, but some detailed information was lacking. A Request for Additional Information (RFI) was sent to PECO by the NRC on September 1, 1981. Thus, this TER is based on the initial submittal and the Licensee's response dated October 7, 1981 (see Appendix C) to the RFI.

4. TECHNICAL EVALUATION

4.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 4.1.3.1.1 requires demonstrating that the SDV drain and vent valves are operable by:

a. verifying each valve to be open at least once per 31 days (valves may be closed intermittently for testing under administrative controls)

b. cycling each valve at least one complete cycle of full travel at least once per 92 days.

LICENSEE RESPONSE

The Licensee proposed revisions of pages 100 and 112 of the Technical Specifications that incorporate SDV drain and vent valves surveillance requirements as follows:

"b. The scram discharge volume drain and vent valves shall be verified open at least once per month. These valves may be closed intermittently for testing.

c. At least once every 3 months verify that the scram discharge volume drain and vent valves closed within 15 seconds after receipt of a closure signal, and reopen upon reset of the closure signal." (Quoted from revised page 100.)

Operability of the scram discharge volume vent and drain valves is necessary for maintaining a reservoir to contain the water exhausted from all control rod drives during a scram." (Quoted from revised page 112.)

In addition, responding to the request for additional information (RFI), the Licensee provided technical bases (given below) for changing the frequency of cycling the drain and vent valves from "at least once per 31 days" (original requirement of paragraph 4.1.3.1.1) to "at least once per 92 days" (present requirement):

"I. Request: Provide technical bases why the licensee proposed surveillance requirement to stroke test the scram discharge volume drain and vent valves every 3 months should not be changed to once per every 31 days.

Response

The Model Technical Specifications, submitted to the licensees in the July 7, 1980 letter requesting an amendment requiring SDV drain and vent valve stroking, specified a 120 day frequency. Philadelphia Electric's proposed amendment specified a more conservative frequency of every 3 months. The Model Technical Specifications, referenced in the September 1, 1981 letter, is a later revision (Fall 1980, revision 3).

A monthly surveillance test would be appropriate for designs lacking redundant valves. However, Philadelphia Electric Company is in the process of adding a second valve in series on each SDV drain and vent line. The modification involves quality assured, environmentally and seismically qualified components. Each valve in series is fed from independent power sources to assure line isolation in the event of a single failure. We believe a stroke test every 3 months is sufficient to ensure isolation capabilities in a redundant valve design. Testing every month will only serve to add to the proliferation of surveillance testing, procedures and paperwork, thus distracting personnel from more essential tasks.

Further justification for the proposed quarterly testing frequency are the permanent modification, described in a letter from S. L. Daltroff to D. G. Eisenhut dated December 16, 1980, that will connect the SDV directly to the instrument volume with new piping equal in cross sectional area. The modifications will provide adequate hydraulic coupling to ensure proper drainage. There will be no dependence on the vent and drain system for the proper detection of water, and additional discharge volume will be provided as added margin for scram capability.

Additionally, monthly testing on a redundant valve design is inconsistent with the testing philosophy presented in the Standard Technical Specifications (Nureg 0123, rev. 3, page 3/4 4-8) for reactor coolant system pressure isolation valves. For example, most primary containment valves are required to be stroke tested only once per 18 months."

FRC EVALUATION

The proposed revision of pages 100 and 112 of the Peach Bottom Atomic Power Station Units 2 and 3 Technical Specifications complies with the requirements of paragraph 4.1.3.1.1 a and b of the NRC staff's Model Technical Specifications regarding surveillance requirements for SDV drain and vent valves.

4.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.1 and Table 3.3.1-1 require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, a total of 4 operable channels containing 4 limit switches per 2 trip systems for the reactor protection system which automatically initiates scram.

Paragraph 3.3.1 and Table 3.3.1-2 concern the response time of the reactor protection system for the functional unit of SDV water level-high which should be specified for each BWR (it is not specified in the table). Paragraph 4.3.1.1 and Table 4.3.1.1-1 require that each reactor protection system instrumentation channel containing a limit switch be shown to be operable for the functional unit of SDV water level-high by the Channel Functional Test monthly and Channel Calibration at each refueling outage. The applicable operational conditions for these requirements are Startup, Run, and Refuel.

LICENSEE RESPONSE

In the submittal of October 14, 1980, the Licensee enclosed the revised page 42, and in a letter dated October 7, 1981 in answer to an RFI, provided the following information (see Appendix C):

> "II. Request: Provide reference to that section of the Technical Specifications which indicates compliance with the following provisions of the Model Technical Specifications.
>
> a. SDV level switch design based on a 1 out of 2 logic.
> b. SDV level switches calibrated every refueling cycle.
>
> <u>Response</u>
>
> Copies of the Peach Bottom Technical Specifications, pages 38, 39, 44, and 46 are enclosed to document compliance with the Model Technical Specifications."

Page 38, Table 3.1.1 (cont'd), "Reactor Protection System (Scram) Instrumentation Requirement," contains the following information for "Trip Function High Water Level in Scram Discharge Volume":

"1. Minimum No. of Operable Instrument Channels per Trip Systems (1): 2

2. Trip Level Setting: ≤ 50 gallons

3. Modes in which Function Must be Operable: Refuel (7)(2), Startup, Run

4. Number of Instrument Channels Provided by Design: 4 Instrument Channels

5. Action (1): A."

Notes for Table 3.1.1 are provided on page 39 and are given below:

"(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 startup 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% of rated.

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

(7) When the reactor is subcritical and the reactor water temperature is less than 212°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 volume high level."

Pages 38 and 39 of the Peach Bottom Technical Specifications address the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.3.1-1 and provide a trip level setting of ≤ 50 gallons.

The requirements of paragraph 3.3.1 and Table 3.3.1-2 are dealt with on page 111 of the Peach Bottom Technical Specifications which gives the reactor protection system response time for Unit 3 as follows (the wording for Unit 2 differs slightly);

> "In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero and approximately 200 milliseconds later, control rod motion begins. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C."

The NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1 are addressed by the revised page 42 and original pages 43, 44, and 46 of the Peach Bottom Technical Specification. The revised page 42 contains Table 4.1.1 (Cont'd), "Reactor Protection System (Scram) Instrument Functional Tests, Minimum Functional Test Frequencies for Safety Instrument and Control Circuits," with the following information for "High Water Level in Scram Discharge Tank":

"1. Group (2): A

2. Functional Test: Trip Channel and Alarm

3. Minimum Frequency (3): Every 1 month."

Notes for Table 4.1.1 from page 43 are:

"2. A description of each of the groups is included in the Bases of this Specification.

3. Functional test are not required on the part of the system that is not required to be operable or are tripped.

   If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

   A. On-off sensors that provide a scram trip function" (from page 51).

The original page 44 contains Table 4.1.2, "Reactor Protection System (Scram) Instrument Calibration, Minimum Calibration Frequencies for Reactor

Protection Instrument Channels," with the following information for "Instrument Channel High Water Level in Scram Discharge Volume":

"1. Group (1): A

2 Calibration (4): Water Column

3. Minimum Frequency (2): Every refueling outage."

From page 46:

"NOTES FOR TABLE 4.1.2

1. A description of three groups is included in the bases of this specification.

2. Calibration test is not required on the part of the system that are not required to be operable or are tripped but is required prior to return to service.

4. Response time is not a part of the routine instrument channel test but will be checked once per operating cycle."

FRC EVALUATION

The Licensee's response to the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.3.1-1 is acceptable. The Peach Bottom Atomic Power Station Units 2 and 3 reactor protection system SDV water level-high instrumentation consists of 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per 2 trip systems, making 1-out-of-2-taken-twice logic. The original page 38 with Table 3.1.1 also specifies $\leq$ 50 gal as a trip setting for scram initiation and applicable operating conditions of Refuel, Startup, and Run, which are acceptable.

The reactor protection system response time of 390 milliseconds specified on original page 111 of the Peach Bottom Technical Specifications addresses the requirements of paragraph 3.3.1 and Table 3.3.1-2 and is acceptable.

The revised page 42 and original pages 43, 44, and 46 of the Peach Bottom Technical Specifications meet the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1. and Table 4.3.1.1-1 that the Channel Functional Test be performed monthly and Channel Calibration at each refueling outage.

4.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.6 and Table 3.3.6-1 require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high, and 1 operable channel containing 1 limit switch for SDV trip bypassed. Paragraph 3.3.6 also requires specifying the trip setpoint for control rod withdrawal block instrumentation monitoring SDV water level-high as indicated in Table 3.3.6-2.

Paragraph 4.3.6 and Table 4.3.6-1 require that each control rod withdrawal block instrumentation channel containing a limit switch be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

LICENSEE RESPONSE

In the submittal dated October 14, 1980, the Licensee proposed to revise pages 73, 74, 83, and 92 of the Peach Bottom Technical Specifications, and in the letter of October 7, 1981, written in response to a RFI, provided this additional information:

"III. Request: Specify "2" minimum operable channels per trip function for the SDV high water level control rod withdrawal block.

<u>Response</u>

We specified "one" minimum operable channel per trip function on page 73 of the proposed amendment because the Peach Bottom design consists of only one channel for the rod block feature associated with high SDV water level.

Six level switches on the scram discharge volume, set at three different water levels, guard against operation of the reactor without sufficient free volume present in the scram discharge volume to receive the scram discharge water in the event of a scram. At the first (lowest) level, one level switch initiates an alarm for operator action. At the second level, another level switch initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, four level switches (two for each Reactor Protection System trip system) initiate a scram to shut down the reactor while sufficient free volume is available to receive the scram discharge.

The modification to the SDV piping, described in our response to item I above, will provide a 40% increase in the discharge volume. This improvement has the effect of substantially increasing the sensitivity of the instrument volume water detection system.

The addition of another level switch to provide a total of two channels for a control rod withdrawal block would have a negligible impact on the probability that the lack of sufficient free volume in the scram discharge volume would go undetected. We believe the current alarm/rod block/scram control circuitry involving six level switches provides the necessary protection.

IV. Request: Provide a technical bases for not providing "scram trip bypassed" instrumentation.

Response

Peach Bottom has the control rod withdrawal block feature when the SDV scram trip is bypassed. A manual keylock switch located in the control room permits the operator to bypass the scram discharge volume high level scram trip if the mode switch is in Shutdown or Refuel. This bypass allows the operator to reset the Reactor Protection System, so that the system is restored to operation while the operator drains the SDV. Additionally, the bypass initiates a control rod block. An annunciator in the control room indicates the bypass condition.

A functional test for the scram trip bypassed-control rod block feature was not proposed in our amendment application due to the simplicity of the design. No relays are involved, only manual switch contacts. Should you consider a functional test to be necessary, we would propose a once per refueling cycle frequency in lieu of the monthly test recommended in the Model Technical Specifications referenced in the July 7, 1980 letter (D. G. Eisenhut to All Operating BWR's). The once per refueling cycle frequency is appropriate for this feature based on its minor safety significance and the simplicity of its design."

The revised pages 73 and 74 of the Peach Bottom Technical Specifications address the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 and Table 3.3.6-1. The revised page 73 contains Table 3.2.C, "Instrumentation That Initiates Control Rod Blocks," with the following information for "Instrument Scram Discharge Volume High Level":

"1. Minimum No. of Operable Instrument Channels Per Trip System: 1

2. Trip Level Setting: $\leq$ 25 gallons

3. Number of Instrument Channels Provided by Design: 1 Inst. Channel

4. Action: (9)."

From the revised page 74 for Units 2 and 3:

"9. If the number of operable channels is less[*] than required by the minimum operable per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn."

The revised page 83 provides Table 4.2.C, "Minimum Test and Calibration Frequency for Control Rod Blocks Actuation," with the following information for "Instrument Channel Scram Discharge Volume-High Level":

"1. Instrument Function Test: Quarterly

2. Calibration: Once/Operating Cycle

3. Instrument Check: NA."

The above information deals with the NRC staff's Model Technical Specification requirements of paragraph 4.3.6 and Table 4.3.6-1. The revised page 92 does not contain any information that would affect the evaluation performed in this report.

FRC EVALUATION

The existing Peach Bottom Atomic Power Station Units 2 and 3 scram discharge system has six level switches on the scram discharge volume (see FSAR, page 3.4-16) set at three different water levels to guard against operation of the reactor without sufficient free volume present in the scram

*Note 9 on page 74 for Unit 3 has a typing error in the first line - "less" is omitted.

discharge headers to receive the scram discharge water in the event of a scram. At the first (lowest) level, one level switch initiates an alarm for operator action. At the second level, with a setpoint of $\leq$ 25 gallons (see revised page 73, Table 3.2.C), one level switch initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, with a setpoint of $\leq$ 50 gallons (see page 38, Table 3.1.1 of the Peach Bottom Technical Specifications), the four level switches (two for each reactor protection system trip system) initiate a scram to shut down the reactor while sufficient free volume is available to receive the scram discharge water.

Reference 9, page, 50, defines Design Criterion 9 ("Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation"), gives the technical basis for "Long-Term Evaluation of Scram Discharge System," and defines acceptable compliance ("The present alarm and rod block instrumentation meets this criterion given adequate hydraulic coupling with the SDV headers"). Thus, if the Peach Bottom Atomic Power Station Units 2 and 3 scram discharge system is modified (long term) so that the hydraulic coupling between scram discharge headers and instrumented volume is adequate and acceptable, then the present alarm and rod block instrumentation, consisting of one operable instrument channel with one limit switch for control rod withdrawal block as specified on the revised page 73, is also acceptable.

In the Peach Bottom Atomic Power Station Units 2 and 3, "Scram Discharge Volume Scram Trips" cannot be bypassed while the reactor is in operational conditions of startup and run (see FSAR, page 7.2-13), and operational condition "refuel with more than one control rod withdrawn" is not applicable because interlocks are provided which prevent the withdrawal of more than one control rod with the mode switch in the refuel position. Thus, the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6, Table 3.3.6-1, paragraph 4.3.6, and Table 4.3.6-1 are not applicable to the Peach Bottom Atomic Power Station for "Trip Function 5b. SDV Scram Trip Bypassed" and were not addressed in the proposed revision of pages 73 and 83. This is acceptable.

The trip setpoint of $\leq$ 25 gallons for control rod withdrawal block instrumentation is acceptable (see revised page 73 of the Peach Bottom Atomic Power Station Units 2 and 3 Technical Specifications). The Licensee's proposed revision of page 83 to meet the requirements of paragraph 4.3.6 and Table 4.3.6-1 is also acceptable since it prescribes a quarterly Channel Functional Test of each control rod withdrawal block instrumentation channel containing a limit switch, and Channel Calibration once per operating cycle for SDV water level-high.

5. CONCLUSIONS

Table 5-1 summarizes the results of the final review and evaluation of the Peach Bottom Atomic Power Station Units 2 and 3 Phase 1 proposed Technical Specifications changes for SDV long-term modification in regard to surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements for reactor protection system and control rod block SDV limit switches. The following conclusions were made:

- The proposed revisions of pages 42, 73, 74, 83, 92, 100, and 112 are acceptable.
- The above revised pages and unrevised pages 38, 39, 43, 44, 46, and 111 meet the surveillance requirements of the NRC staff's Model Technical Specifications.

Table 5-1 Evaluation of Phase 1 Proposed Technical Specifications Changes for Scram Discharge Volume Long-Term Modifications Peach Bottom Atomic Power Station Units 2 and 3

Surveillance Requirements	Technical Specifications: NRC Staff Model (Paragraph)	Technical Specifications: Proposed by Licensee	Evaluation
SDV DRAIN AND VENT VALVES			
Verify each valve open	Once per 31 days (4.1.3.1.1a)	Once per month (pp. 100 & 112, revised)	Acceptable
Cycle each valve one complete cycle	Once per 92 days (4.1.3.1.1b)	Every 3 months (pp. 100 & 112, revised)	Acceptable
REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES			
Minimum operable channels per trip system	2 (3.3.1, Table 3.3.1-1)	2 (pp. 38 & 39, Table 3.1.1)	Acceptable
SDV water level-high response time	NA (3.3.1, Table 3.3.1-2)	0.390 sec. max. (p. 111)	Acceptable
SDV water level-high			
Channel functional test	Monthly (4.3.1.1, Table 4.3.1.1-1)	Every 1 month (pp. 42 & 43, Table 4.1.1, revised)	Acceptable
Channel calibration	Each refueling (4.3.1.1, Table 4.3.1.1-1)	Each refueling (pp. 44 & 46, Table 4.1.2)	Acceptable

Table 5-1 (Cont.)

Surveillance Requirements	Technical Specifications: NRC Staff Model (Paragraph)	Technical Specifications: Proposed by Licensee	Evaluation
CONTROL ROD BLOCK SDV LIMIT SWITCHES			
Minimum operable channels per trip function			
SDV water level-high	2 (3.3.6, Table 3.3.6-1)	1 (pp. 73 & 74, Table 3.2.C, revised)	Acceptable*
SDV scram trip bypassed	1 (3.3.6, Table 3.3.6-1)	NA (pp. 73 & 74, Table 3.2.C, revised)	Acceptable*
SDV water level-high			
Trip setpoint	NA (3.3.6, Table 3.3.6-2)	$\leq$ 25 gallons (pp. 73 & 74, Table 3.2.C, revised)	Acceptable
Channel functional test	Quarterly (4.3.6, Table 4.3.6-1)	Quarterly (p. 83, Table 4.2.C, revised)	Acceptable
Channel calibration	Each refueling (4.3.6, Table 4.3.6-1)	Once per operating cycle (p. 83, Table 4.2.C, revised)	Acceptable
SDV scram trip bypassed			
Channel functional test	Monthly (4.3.6, Table 4.3.6-1)	NA	Acceptable*

* See Reference 9, p. 50, and pp. 22 and 23 of this TER.

6. REFERENCES

1. IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capacity"
   NRC, Office of Inspection and Enforcement, June 12, 1980

2. D. G. Eisenhut (NRR), letter "To All Operating Boiling Water Reactors (BWRs)" with enclosure, "Model Technical Specifications"
   July 7, 1980

3. IE Bulletin 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 3, 1980

4. IE Bulletin 80-17, Supplement 1, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 18, 1980

5. IE Bulletin 80-17, Supplement 2, "Failures Revealed by Testing Subsequent to Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 22, 1980

6. IE Bulletin 80-17, Supplement 3, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, August 22, 1980

7. IE Bulletin 80-17, Supplement 4, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, December 18, 1980

8. IE Bulletin 80-17, Supplement 5, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, February 13, 1981

9. P. S. Check (NRR), memorandum with enclosure, "Generic Safety Evaluation Report BWR Scram Discharge System"
   December 1, 1980

10. P. S. Check (NRR), memorandum with enclosure, "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17"
    June 10, 1981

APPENDIX A

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS*

* Note: Applicable changes are marked by vertical lines in the margins.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days and

b. Cycling each valve through at least one complete cycle of full travel at least once per 92 days.

4.1.3.1.2 When above the preset power level of the RWM and RSCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

*These valves may be closed intermittently for testing under administrative controls.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position (6), based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed (7.0) seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding (7.0) seconds:

a. Declare the control rod(s) with the slow insertion time inoperable, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of (7.0) seconds, or

c. Be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one inoperable channel in at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

* If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
8. Scram Discharge Volume Water Level - High	1, 2, $5^{(h)}$	2	4
9. Turbine Stop Valve - Closure	$1^{(i)}$	$4^{(j)}$	7
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$1^{(i)}$	$2^{(j)}$	7
11. Reactor Mode Switch in Shutdown Position	1, 2, 3, 4, 5	1	8
12. Manual Scram	1, 2, 3, 4, 5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to < (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER, within 2 hours..

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, verify all insertable control rods to be fully inserted within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 9 In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than (11) LPRM inputs to an APRM channel.

(d) These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) These functions are automatically bypassed when turbine first stage pressure is ≤ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - Upscale	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, (15)%	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ (0.09)**
c. Fixed Neutron Flux - Upscale, (118)%	≤ (0.09)
d. Inoperative	NA
e. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ (0.55)
4. Reactor Vessel Water Level - Low, Level 3	≤ (1.05)
5. Main Steam Line Isolation Valve - Closure	≤ (0.06)
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	≤ (0.06)
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ (0.08)#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Trip Oil Pressure - Low	NA	M	Q	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) Within 24 hours prior to startup, if not performed within the previous 7 days.
(c) The IRM and SRM channels shall be determined to overlap for at least ( ) decades during each startup and the IRM and APRM channels shall be determined to overlap for at least ( ) decades during each controlled shutdown, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM readout to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

INSTRUMENTATION

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[a]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Simulated Thermal Power - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale[c]	3	2	61
	2	5	61
c. Inoperative[c]	3	2	61
	2	5	61
d. Downscale[d]	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in (e)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale[e]	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Trip Bypassed	1	(1, 2, 5**)	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. (Comparator) (Downscale)	2	1	62

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TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With THERMAL POWER $\geq$ (20)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (2) or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	$\leq$ 0.66 W + (40)%	$\leq$ 0.66 W + (43)%
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5)% of RATED THERMAL POWER	$\geq$ (3)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power - Upscale	$\leq$ 0.66 W + (42)%*	$\leq$ 0.66 W + (45)%*
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5)% of RATED THERMAL POWER	$\geq$ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ (12)% of RATED THERMAL POWER	$\leq$ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ (2 x 10^5) cps	$\leq$ (5 x 10^5) cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ (3) cps	$\geq$ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ (108/125) of full scale	$\leq$ (110/125) of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ (5/125) of full scale	$\geq$ (3/125) of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level High	To be specified	NA
b. Scram Trip Bypassed	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ (__/__) of full scale	$\leq$ (__/__) of full scale
b. Inoperative	NA	NA
c. (Comparator) (Downscale)	$\leq$ (10)% flow deviation	$\leq$ (__)% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[b], M	Q	1*
b. Inoperative	NA	S/U[b], M	NA	1*
c. Downscale	NA	S/U[b], M	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power - Upscale	NA	S/U[b], M	Q	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5
c. Downscale	NA	S/U[b], M	Q	1
d. Neutron Flux - Upscale, Startup	NA	S/U[b], M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[b], W[c]	NA	2, 5
b. Upscale	NA	S/U[b], W[c]	Q	2, 5
c. Inoperative	NA	S/U[b], W[c]	NA	2, 5
d. Downscale	NA	S/U[b], W[c]	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b], W[c]	NA	2, 5
b. Upscale	NA	S/U[b], W[c]	Q	2, 5
c. Inoperative	NA	S/U[b], W[c]	NA	2, 5
d. Downscale	NA	S/U[b], W[c]	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[b], M	Q	1
b. Inoperative	NA	S/U[b], M	NA	1
c. (Comparator) (Downscale)	NA	S/U[b], M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. When making an unscheduled change from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2.

* With THERMAL POWER ≥ (20)% of RATED THERMAL POWER.

** With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

APPENDIX B

PHILADELPHIA ELECTRIC COMPANY LETTER OF OCTOBER 14, 1980

AND

SUBMITTAL WITH PROPOSED TECHNICAL SPECIFICATIONS CHANGES

FOR

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

PHILADELPHIA ELECTRIC COMPANY

2301 MARKET STREET

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ASSISTANT COUNSEL

October 14, 1980

Dr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

Re: Peach Bottom Atomic Power Station Units 2 and 3
Docket Nos. 50-277 and 50-278

Dear Dr. Denton:

Enclosed for filing with the Commission are three originals and 37 copies of Philadelphia Electric Company's Application for Amendment of Facility Operating Licenses DPR-44 and DPR-56. This Application requests the addition of surveillance requirements for the Scram Discharge Volume (SDV) vent and drain valves and Limiting Condition for Operation/surveillance requirements for Reactor Protection System and control rod block SDV limit switches, as requested by Mr. D. G. Eisenhut in his letter of July 7, 1980.

Pursuant to Section 170.12 of the Commission's regulations, there is enclosed a check payable to the United States Nuclear Regulatory Commission in the amount of $4,400 to cover the filing fee for this Application.

Very truly yours,

Eugene J. Bradley

A001
5/40
w/check
$4,400.00

EJB:mk
Enclosures

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Dupe 8010200151

BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	:	
	:	Docket Nos. 50-277
PHILADELPHIA ELECTRIC COMPANY	:	50-278

APPLICATION FOR AMENDMENT

OF

FACILITY OPERATING LICENSES

DPR-44 & DPR-56

Edward G. Bauer, Jr.
Eugene J. Bradley

2301 Market Street
Philadelphia, Pennsylvania 19101

Attorneys for
Philadelphia Electric Company

8610200459

BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	:	
	:	Docket Nos. 50-277
PHILADELPHIA ELECTRIC COMPANY	:	50-278

APPLICATION FOR AMENDMENT

OF

FACILITY OPERATING LICENSES

DPR-44 & DPR-56

Philadelphia Electric Company, Licensee under Facility Operating Licenses DPR-44 and DPR-56 for Peach Bottom Units 2 and 3, hereby requests that the Technical Specifications contained in Appendix A of the Operating Licenses be amended by revising certain sections as indicated by a vertical bar in the margin of attached pages 42, 73, 74, 83, 92, 100, and 112.

Correspondence from Mr. D. G. Eisenhut, Director, Division of Licensing, NRC, to All Boiling Water Reactors, dated July 7, 1980, requested each Licensee to submit a license amendment application incorporating surveillance requirements for the Scram Discharge Volume (SDV) vent and drain valves and

Limiting Condition for Operation/surveillance requirements for Reactor Protection System and control rod block SDV limit switches. The licensee proposes an amendment that is consistent with the Model Technical Specifications submitted with Mr. Eisenhut's letter, except where changes were necessary to reflect the design of the Peach Bottom scram discharge volume controls. Operability and surveillance requirements for the scram trip bypass control rod withdrawal block was not proposed for incorporation into this license amendment application since sufficient supervisory instrumentation is available for monitoring the status of the scram bypass feature. The supervisory instrumentation includes a "Scram Discharge Volume High Water Level Scram Bypassed" annunciator in the control room.

The Licensee proposes revisions to pages 100 and 112 of the Technical Specifications that incorporate SDV drain and vent valve surveillance requirements. These changes conform with the Model Technical Specification guidelines except where necessary to reflect the following features of the Peach Bottom design: (1) operability of the control rods is not dependent on the operability of the SDV drain and vent valves, and (2) individual control rod scram testing does not actuate the SDV drain and vent valves. Only a full reactor scram actuates these valves. A testing procedure that verifies operability of these valves and does not involve scram testing, will be performed quarterly.

The Peach Bottom Technical Specifications, pages 38 and 39 currently specify operability requirements for the Reactor Protection System SDV high level trip that parallel the Model

Technical Specification requirements. The Licensee proposes revisions to the Reactor Protection System SDV high level trip surveillance requirements (page 42) that conform with the Model Technical Specifications.

The Licensee proposes revisions to the Technical Specifications (pages 73, 74, 83 and 92) that incorporate operability and surveillance requirements for the Control Rod Withdrawal Block SDV high level trip that are consistent with the Model Technical Specifications.

The Licensee proposes that the reference to a specific MCPR limit (1.07) on page 92 (Unit 3) be replaced with "the fuel cladding integrity safety limit" to bring this page into conformity with page 92 (Unit 2).

Pursuant to 10 CFR 170.22, "Schedule of Fees for Facility License Amendments", Philadelphia Electric Company proposes that this Application for Amendment be considered a Class III Amendment for Unit 2, and a Class I Amendment for Unit 3, since the proposed changes have acceptability for the issue clearly identified by an NRC position, and are deemed not to involve a significant hazards consideration.

The Plant Operation Review Committee and the Operation and Safety Review Committee have reviewed these proposed changes to the Technical Specifications, and have concluded that they do not involve an unreviewed safety question or a significant hazards consideration, and will not endanger the health and safety of the public.

Respectfully submitted,

Vice President

COMMONWEALTH OF PENNSYLVANIA :
ss.
COUNTY OF PHILADELPHIA :

S. L. Daltroff, being first duly sworn, deposes and says:

That he is Vice President of Philadelphia Electric Company, the Applicant herein; that he has read the foregoing Application for Amendment of Facility Operating Licenses and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

S. L. Daltroff

Subscribed and sworn to
before me this 10th day
of October, 1980

Elizabeth H. Boyer
Notary Public

ELIZABETH H. BOYER
Notary Public, Phila., Phila. Co.
My Commission Expires Jan. 30, 1982

CERTIFICATE OF SERVICE

I certify that service of the foregoing Application was made upon the Board of Supervisors, Peach Bottom Township, York County, Pennsylvania, by mailing a copy thereof, via first-class mail, to Albert R. Steele, Chairman of the Board of Supervisors, R. D. No. 1, Delta, Pennsylvania 17314; upon the Board of Supervisors, Fulton Township, Lancaster County, Pennsylvania, by mailing a copy thereof, via first-class mail, to George K. Brinton, Chairman of the Board of Supervisors, Peach Bottom, Pennsylvania 17563; and upon the Board of Supervisors, Drumore Township, Lancaster County, Pennsylvania, by mailing a copy thereof, via first-class mail, to Wilmer P. Bolton, Chairman of the Board of Supervisors, R. D. No. 1, Holtwood, Pennsylvania 17532; all this 14th day of October, 1980.

Eugene J. Bradley
Attorney for
Philadelphia Electric Company

TABLE 4.1.1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	GROUP (2)	FUNCTIONAL TEST	MINIMUM FREQUENCY (3
High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Every 1 month
Turbine Condenser Low Vacuum (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1)
Main Steam Line High Radiation	B1	Trip Channel and Alarm (4)	Once/week
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Every 1 month (1)
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Every 1 month
Turbine First Stage Pressure Permissive	A	Trip Channel and Alarm	Every 3 months (1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Every 1 month (1)
Reactor Pressure Permissive (6)	B2	Trip Channel and Alarm (4)	Every 3 months.

TABLE 4.1.1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	GROUP (2)	FUNCTIONAL TEST	MINIMUM FREQUENCY (3)
High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Every 1 month
Turbine Condenser Low Vacuum (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1)
Main Steam Line High Radiation	B1	Trip Channel and Alarm (4)	Once/week
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Every 1 month (1)
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Every 1 month
Turbine First Stage Pressure Permissive	A	Trip Channel and Alarm	Every 3 months (1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Every 1 month (1)
*Reactor Pressure Permissive (6)	B2	Trip Channel and Alarm (4)	Every 3 months.
**Reactor Pressure Permissive	A	Trip Channel and Alarm	Every 3 months

* Deleted when modification authorized by Amendment No. 67 are completed.
** Effective when modifications authorized by Amendment No. 67 are completed.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq(0.66W+42)\times\frac{FRP}{MFLPD}$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Startup Mode)	$\leq$12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq(0.66W+41)\times\frac{FRP}{MFLPD}$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (3)	$\geq$2.5 indicated on scale	8 Inst. Channels	(1)
3	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$108 indicated on scale	8 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Volume High Level	$\leq$ 25 gallons	1 Inst. Channel	(9)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first colum cannot be met for both trip systems, the systems shall be tripped.

2. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP) where:

   FRP = fraction of rated thermal power (3293 MWt)

   MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for all 8x8 fuel.

   The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

   W = Loop Recirculation flow in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

   Trip level setting is in percent of rated power (3293 MWt).

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function is bypassed when the IRM range switches are on range 8 or above.

7. The trip is bypassed when the reactor power is $\leq$ 30%.

8. This function is bypassed when the mode switch is placed in Run.

9. If the number of operable channels is less than required by the minimum operable per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn.

PBAPS

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP) where:

   FRP = fraction of rated thermal power (3293 MWt)

   MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for all 7x7 fuel and 13.4 KW/ft for all 8x8 fuel.

   The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

   W = Loop Recirculation flow in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

   Trip level setting is in percent of rated power (3293 MWt).

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function is bypassed when the IRM range switches are on range 8 or above.

7. The trip is bypassed when the reactor power is $\leq$ 30%.

8. This function is bypassed when the mode switch is placed in Run.

9. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, "Startup" mode and "Refuel" mode if more than one control rod is withdrawn.

Franklin Research Center
A Division of The Franklin Institute

B-14

Table 4.2.C

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

	Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
1)	APRM - Downscale	(1) (3)	Once/3 months	Once/day
2)	APRM - Upscale	(1) (3)	Once/3 months	Once/day
3)	IRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
4)	IRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
5)	RBM - Upscale	(1) (3)	Once/6 months	Once/day
6)	RBM - Downscale	(1) (3)	Once/6 months	Once/day
7)	SRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
8)	SRM - Detector Not in Startup Position	(2) (3)	Startup or Control Shutdown	(2)
9)	IRM - Detector Not in Startup Position	(2) (3)	Startup or Control Shutdown	(2)
10)	Scram Discharge Volume - High Level	Quarterly	Once/Operating Cycle	NA

Logic System Functional Test (4) (6)	Frequency
(1) System Logic Check	Once/6 months

3.2 BASES (Cont'd)
The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequences. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit.

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in the local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in the control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

High water level in the scram discharge volume may be indicative of excessive scram valve leakage, or plugging or closing of the discharge volume drain valve, and could jeopardize the ability of all rods to fully insert on a scram signal.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function (ADS) is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale

LIMITING CONDITIONS FOR OPERATION

3.3.A Reactivity Limitations (Cont'd)

failure is not due to a failed control rod drive mechanism collet housing.

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.

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.

SURVEILLANCE REQUIREMENTS

4.3.A Reactivity Limitations (Cont'd)

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 scram discharge volume drain and vent valves shall be verified open at least once per month. These valves may be closed intermittently for testing.

c. At least once every 3 months verify that the scram discharge volume drain and vent valves closed within 15 seconds after receipt of a closure signal, and reopen upon reset of the closure signal.

d. 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.

PBAPS

3.3 & 4.3 BASES (Cont'd)

identified as the resistance to drive motion by an internal control rod drive filter. The filter had been loaded by foreign material, probably accelerated by construction debris. The sudden changes in drive scram performance which were observed at that plant were due to stepwise release into reactor coolant of particulate matter as the reactor and subsystem were subsequently started up. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Dresden 2 has currently 27 "B" type drives. Approximately 4718 drive tests have been recorded to date. Data documenting the successful performance of the modified drive has been submitted to the NRC with a letter from Commonwealth Edison Company to the Commission dated November 6, 1972 with the subject of the letter being Proposed Changes to Quad-Cities Power Station Operating License, including Appendices A and B, DPR 29 and 30, AEC Dkts 50-254 and 50-265.

Although the cause and cure of the dirt problem were known at the time of the writing of the Dresden 3 Tech Specs, the progressive surveillance requirement was incorporated into the technical specification to ostensibly detect any other unforeseen drive problems. The possibility of this being a temporary requirement may be inferred from the provision for review of all surveillance requirements after the first operating cycle.

Operability of the scram discharge volume vent and drain valves is necessary for maintaining a reservoir to contain the water exhausted from all control rod drives during a scram.

APPENDIX C

PHILADELPHIA ELECTRIC COMPANY LETTER OF OCTOBER 7, 1981

WITH

RESPONSE TO RFI REGARDING

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

COPY

PHILADELFHIA ELECTRIC COMPANY
2301 MARKET STREET
P.O. BOX 8699
PHILADELPHIA, PA. 19101
(215) 841-5001

1881 - 1981

SHIELDS L. DALTROFF
VICE PRESIDENT
ELECTRIC PRODUCTION

October 7, 1981

Re: Docket Nos. 50-277
50-278

Mr. John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Correspondence dated September 1, 1981,
J. F. Stolz, NRC to E. G. Bauer, Jr.,
Philadelphia Electric Company

Dear Mr. Stolz:

This letter provides the information you requested for your contractor in the referenced letter regarding Philadelphia Electric Company's application for amendment to the Peach Bottom Atomic Power Station Technical Specification. The application was submitted on October 14, 1980, at the request of the NRC to provide surveillance requirements for scram discharge volume (SDV) vent and drain valves and LCO/surveillance requirements for RPS and control rod block SDV limit switches. The requests and our responses are provided sequentially as follows:

I. Request: Provide technical bases why the licensee proposed surveillance requirement to stroke test the scram discharge volume drain and vent valves every 3 months should not be changed to once every 31 days.

Duyse
~~8110150348~~

Response

The Model Technical Specifications, submitted to the licensees in the July 7, 1980 letter requesting an amendment requiring SDV drain and vent valve stroking, specified a 120 day frequency. Philadelphia Electric's proposed amendment specified a more conservative frequency of every 3 months. The Model Technical Specifications, referenced in the September 1, 1981 letter, is a later revision (Fall 1980, revision 3).

A monthly surveillance test would be appropriate for designs lacking redundant valves. However, Philadelphia Electric Company is in the process of adding a second valve in series on each SDV drain and vent line. The modification involves quality assured, environmentally and seismically qualified components. Each valve in series is fed from independent power sources to assure line isolation in the event of a single failure. We believe a stroke test every 3 months is sufficient to ensure isolation capabilities in a redundant valve design. Testing every month will only serve to add to the proliferation of surveillance testing, procedures and paperwork, thus distracting personnel from more essential tasks.

Further justification for the proposed quarterly testing frequency are the permanent modifications, described in a letter from S. L. Daltroff to D. G. Eisenhut dated December 16, 1980, that will connect the SDV directly to the instrument volume with new piping equal in cross sectional area. The modifications will provide adequate hydraulic coupling to ensure proper drainage. There will be no dependence on the vent and drain system for the proper detection of water, and additional discharge volume will be provided as added margin for scram capability.

Additionally, monthly testing on a redundant valve design is inconsistent with the testing philosophy presented in the Standard Technical Specifications (Nureg 0123, rev. 3, page 3/4 4-8) for reactor coolant system pressure isolation valves. For example, most primary containment valves are required to be stroke tested only once per 18 months.

II. Request: Provide reference to that section of the Technical Specifications which indicates compliance with the following provisions of the Model Technical Specifications.

   a. SDV level switch design based on a 1 out of 2 logic.
   b. SDV level switches calibrated every refueling cycle.

Response

Copies of the Peach Bottom Technical Specifications, pages 38, 39, 44, and 46 are enclosed to document compliance with the Model Technical Specifications.

III. Request: Specify "2" minimum operable channels per trip function for the SDV high water level control rod with drawal block.

Response

We specified "one" minimum operable channel per trip function on page 73 of the proposed amendment because the Peach Bottom design consists of only one channel for the rod block feature associated with high SDV water level.

Six level switches on the scram discharge volume, set at three different water levels, guard against operation of the reactor without sufficient free volume present in the scram discharge volume to receive the scram discharge water in the event of a scram. At the first (lowest) level, one level switch initiates an alarm for operator action. At the second level, another level switch initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, four level switches (two for each Reactor Protection System trip system) initiate a scram to shut down the reactor while sufficient free volume is available to receive the scram discharge.

The modification to the SDV piping, described in our response to item I above, will provide a 40% increase in the discharge volume. This improvement has the effect of substantially increasing the sensitivity of the instrument volume water detection system.

The addition of another level switch to provide a total of two channels for a control rod withdrawal block would have a negligible impact on the probability that the lack of sufficient free volume in the scram discharge volume would go undetected. We believe the current alarm/rod block/scram control circuitry involving six level switches provides the necessary protection.

IV. Request: Provide a technical bases for not providing "scram trip bypassed" instrumentation.

Response

Peach Bottom has the control rod withdrawal block feature when the SDV scram trip is bypassed. A manual keylock switch located in the control room permits the operator to bypass the scram discharge volume high level scram trip if the mode switch is in Shutdown or Refuel. This bypass allows the operator to reset the Reactor Protection System, so that the system is restored to operation while the operator drains the SDV. Additionally, the bypass initiates a control rod block. An annunciator in the control room indicates the bypass condition.

A functional test for the scram trip bypassed-control rod block feature was not proposed in our amendment application due to the simplicity of the design. No relays are involved, only manual switch contacts. Should you consider a functional test to be necessary, we would propose a once per refueling cycle frequency in lieu of the monthly test recommended in the Model Technical Specifications referenced in the July 7, 1980 letter (D. G. Eisenhut to All Operating BWR's). The once per refueling cycle frequency is appropriate for this feature based on its minor safety significance and the simplicity of its design.

If you have any questions regarding the above or need additional information regarding the Technical Specification on the SDV control systems, please contact William Birely, (215) 841-5048.

Very truly yours,

Original signed by
S. L. DALTROFF

WCB:bas

Enclosure

bcc: V. S. Boyer
J. S. Kemper
J. W. Gallagher
E. J. Bradley
M. J. Cooney
R. H. Moore
W. T. Ullrich
W. M. Alden/W. C. Birely
Franklin Research Center

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 88 and 88 to Facility Operating Licenses 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 Technical Specifications (TSs) for operation of the Peach Bottom Atomic Power Station, Units Nos. 2 and 3 (the facility) located in York County, Pennsylvania. The amendments are effective as of the date of issuance.

The amendments revise the TSs to provide limiting conditions for operation and surveillance requirements for Scram Discharge Volume (SDV) vent and drain valves and reactor protection system and control rod block SDV limit switches.

The application for the amendments 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) the application for amendments dated October 14, 1980, as supplemented October 7, 1981, (2) Amendment No. 88 to License No. DPR-44, and Amendment No. 88 to License No. DPR-56, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania. 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 Licensing.

Dated at Bethesda, Maryland, this 1st day of March 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing