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MAY 5 1980

Docket 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:

The Commission has issued the enclosed Amendment Nos. 68 and 67 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Unit Nos. 2 and 3. These amendments revise the Technical Specifications in response to your application of August 27, 1979, as supplemented by your letters of November 5, 1979, January 30, 1980, February 13, 1980 and March 27, 1980.

The revised Technical Specifications authorize replacement of existing pressure switches that sense drywell and reactor pressures with analog loops and modify two reactor water level indication loops to improve the reliability, accuracy and response time of this instrumentation.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 68 to DPR-44
2. Amendment No. 67 to DPR-56
3. Safety Evaluation
4. Notice

cc w/encl:
See next page

8005220305

OFFICE	DOR:ORB#3	DOR:ORB#3	DOR:ORP	OELD	DOR:ORB#3
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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. 68
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 August 27, 1979, as supplemented by letters dated November 5, 1979, January 30, 1980, February 13, 1980 and March 27, 1980, 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 CRF Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 68 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8005220309 P

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 68

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

1. 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
41	41
42	42
44	44
45	45
*47/48	*47/48
65	65
66	66
81	81
*85/86	*85/86

* Overleaf page, provided for convenience

TABLE 4.1.1

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

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each refueling outage.
Manual Scram	A	Trip Channel and Alarm	Every 3 months.
RPS Channel Test Switch	A	Trip Channel and Alarm	Every refueling outage or after channel maintenance.
IRM			
High Flux	C	Trip Channel and Alarm (4)	One per week during refueling or startup and before each startup.
Inoperative	C	Trip Channel and Alarm (4)	Once per week during refueling or startup and before each startup.
APRM			
High Flux	B1	Trip Output Relays (4)	Once/week.
Inoperative	B1	Trip Output Relays (4)	Once/week.
Downscale	B1	Trip Output Relays (4)	Once/week.
Flow Bias	B1	Calibrate Flow Bias Signal (4)	Once/month (1).
High Flux in Startup or Refuel	C	Trip Output Relays (4)	Once per week during refueling or startup and before each startup.
High Reactor Pressure (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1).
High Drywell Pressure (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1).
Reactor Low Water Level (5) (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1).

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 3 months.
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 4.1.2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdown	Maximum frequency once per week.
APRM High Flux			
Output Signal	B1	Heat Balance	Twice per week.
Flow Bias Signal	B1	With Standard Pressure Source	Every refueling outage.
LPRM Signal	B1	TIP System Traverse	Every 6 weeks.
High Reactor Pressure	B2	Standard Pressure Source	Once per operating cycle.
High Drywell Pressure	B2	Standard Pressure Source	Once per operating cycle.
Reactor Low Water Level	B2	Pressure Standard	Once per operating cycle.
High Water Level in Scram Discharge Volume	A	Water Column	Every refueling outage.
Turbine Condenser Low Vacuum	B2	Standard Vacuum Source	Once per operating cycle.
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5).
Main Steam Line High Radiation	B1	Standard Current Source (3)	Every 3 months.
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 months.

TABLE 4.1.2 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency (2)
Turbine Control Valve Fast Closure Oil Pressure Trip	A	Standard Pressure Source	Once per operating cycle.
Turbine Stop Valve Closure	A	Note (5)	Note (5).
Reactor Pressure Premissive	B2	Standard Pressure Source	Once per operating Cycle.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertant criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference subsection 7.2 FSAR). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

The APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other subchannel. APRM's B, D and F are arranged similarly in

3.0 BASES (Cont'd)

the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, generator load rejection, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

Instrumentation sensing drywell pressure is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to limit fission product release so that 10 CFM Part 100 guidelines are not exceeded. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation of the main condenser off-gas line.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (High flux in Start-up or Refuel) system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges.

The IRM system provides protection against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2	Reactor High Water Level	≤+45 in. indicated level	2 Inst. Channels	Trips HPCI and RCIC turbines.
1	Reactor Low Level (inside shroud)	≥+312 in. above vessel zero (2/3 core height)	2 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition.
2	Containment High Pressure	1 < p < 2 psig	4 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition.
1	Confirmatory Low Level	≥+6 in. indicated level	2 Inst. Channels	ADS Permissive
2	High Drywell Pressure	≤2 psig	4 Inst. Channels	1. Initiates Core Spray; LPCI; HPCI 2. Initiates starting of Diesel Generators 3. Initiates Auto Blowdown (ADS) in conjunction with Low-Low Reactor Water Level, 120 second time delay, and LPCI or Core Spray pump running.

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2	Reactor Low Pressure	400-500 psig	4 Inst. Channels	Permissive for opening Core Spray and LPCI Admission valves. Coincident with high dry well pressure, starts LPCI and Core Spray pumps.
2	Reactor Low Pressure	200-250 psig	4 Inst. Channels	Permissive for closing Recirculating Pump Discharge Valve.
1	Reactor Low Pressure	50≤P≤75 psig	2 Inst. Channels	In conjunction with PCI signal permits closure of RHR (LPCI) injection valves.

TABLE 4.2.B

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

	Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1)	Reactor Water Level (7)	(1) (3)	Once/operating cycle	Once/day
2)	Drywell Pressure (7)	(1) (3)	Once/operating cycle	Once/day
3)	Reactor Pressure (7)	(1) (3)	Once/operating cycle	Once/day
4)	Auto Sequencing Timers	NA	Once/operating cycle	None
5)	ADS - LPCI or CS Pump Disch. Pressure Interlock	(1)	Once/3 months	None
6)	Trip System Bus Power Monitors	(1)	NA	None
7)	Core Spray Sparger d/p	(1)	Once/6 months	Once/day
8)	Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
9)	Steam Line High Temp. (HPCI & RCIC)	(1) (3)	Once/operating cycle	Once/day
10)	Safeguards Area High Temp.	(1)	Once/3 months	None
11)	HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None
12)	HPCI Suction Source Levels	(1)	Once/3 months	None
13)	4KV Emergency Power System Voltage Relays	Once/operating cycle	Once/5 year	None
14)	ADS Relief Valves Bellows Pressure Switches	Once/operating cycle	Once/operating cycle	None
15)	LPCI/Cross Connect Valve Position	Once/refueling cycle	N/A	N/A

TABLE 4.2.E

MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Equipment Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
2) Floor Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
3) Air Sampling System	(1)	Once/3 months	Once/day

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Level	Once/operating cycle	Once Each Shift
2) Reactor Pressure	Once/6 months	Once Each Shift
3) Drywell Pressure	Once/6 months	Once Each Shift
4) Drywell Temperature	Once/6 months	Once Each Shift
5) Suppression Chamber Temperature	Once/6 months	Once Each Shift
6) Suppression Chamber Water Level	Once/6 months	Once Each Shift
7) Control Rod Position	NA	Once Each Shift
8) Neutron Monitoring (APRM)	Twice Per Week	Once Each Shift

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. 67
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 August 27, 1979, as supplemented by letters dated November 5, 1979, January 30, 1980, February 13, 1980 and March 27, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-56 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 67 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 67

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

1. 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
41	41
42	42
44	44
45	45
*47/48	*47/48
65	65
66	66
81	81
*85/86	*85/86

* Overleaf page, provided for convenience

* deleted when modification authorized by Amendment No. are completed

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

TABLE 4.1.1

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

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each refueling outage.
Manual Scram	A	Trip Channel and Alarm	Every 3 months.
RPS Channel Test Switch	A	Trip Channel and Alarm	Every refueling outage or after channel maintenance.
IRM			
High Flux	C	Trip Channel and Alarm (4)	One per week during refueling or startup and before each startup.
Inoperative	C	Trip Channel and Alarm (4)	Once per week during refueling or startup and before each startup.
APRM			
High Flux	B1	Trip Output Relays (4)	Once/week.
Inoperative	B1	Trip Output Relays (4)	Once/week.
Downscale	B1	Trip Output Relays (4)	Once/week.
Flow Bias	B1	Calibrate Flow Bias Signal (4)	Once/month (1).
High Flux in Startup or Refuel	C	Trip Output Relays (4)	Once per week during refueling or startup and before each startup.
High Reactor Pressure (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1).
*High Drywell Pressure	A	Trip Channel and Alarm	Every 1 month (1).
**High Drywell Pressure (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1).
Reactor Low Water Level (5) (6)	B2	Trip Channel and Alarm (4)	Every 1 month (1).

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 3 months.
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. are completed.

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

Amendment No. 67

TABLE 4.1.2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdown	Maximum frequency once per week.
APRM High Flux			
Output Signal	B1	Heat Balance	Twice per week.
Flow Bias Signal	B1	With Standard Pressure Source	Every refueling outage.
LPRM Signal	B1	TIP System Traverse	Every 6 weeks.
High Reactor Pressure	B2	Standard Pressure Source	Once per operating cycle.
*High Drywell Pressure	A	Standard Pressure Source	Once per operating cycle.
**High Drywell Pressure	B2	Standard Pressure Source	Once per operating cycle.
Reactor Low Water Level	B2	Pressure Standard	Once per operating cycle.
High Water Level in Scram Discharge Volume	A	Water Column	Every refueling outage.
Turbine Condenser Low Vacuum	B2	Standard Vacuum Source	Once per operating cycle.
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5).
Main Steam Line High Radiation	B1	Standard Current Source (3)	Every 3 months.
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 months.

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

TABLE 4.1.2 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency (2)
Turbine Control Valve Fast Closure Oil Pressure Trip	A	Standard Pressure Source	Once per operating cycle.
Turbine Stop Valve Closure	A	Note (5)	Note (5).
**Reactor Pressure Permissive	B2	Standard Pressure Source	Once per operating Cycle.
*Reactor Pressure Permissive	A	Standard Pressure Source	Every 6 months

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

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertant criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference subsection 7.2 FSAR). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

The APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other subchannel. APRM's B, D and F are arranged similarly in

3.0 BASES (Cont'd)

the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, generator load rejection, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

Instrumentation sensing drywell pressure is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to limit fission product release so that 10 CFM Part 100 guidelines are not exceeded. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation of the main condenser off-gas line.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (High flux in Start-up or Refuel) system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges.

The IRM system provides protection against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2	Reactor High Water Level	≤+45 in. indicated level	2 Inst. Channels	Trips HPCI and RCIC turbines.
1	Reactor Low Level (inside shroud)	≥+312 in. above vessel zero (2/3 core height)	2 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition.
2	Containment High Pressure	1 < p < 2 psig	4 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition.
1	Confirmatory Low Level	≥+6 in. indicated level	2 Inst. Channels	ADS Permissive
2	High Drywell Pressure	≤2 psig	4 Inst. Channels	1. Initiates Core Spray; LPCI; HPCI 2. Initiates starting of Diesel Generators 3. Initiates Auto Blowdown (ADS) in conjunction with Low-Low Reactor Water Level, 120 second time delay, and LPCI or Core Spray pump running.

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
2	Reactor Low Pressure	400-500 psig	4 Inst. Channels	Permissive for opening Core Spray and LPCI Admission valves. Coincident with high dry well pressure, starts LPCI and Core Spray pumps.
2	Reactor Low Pressure	200-250 psig	4 Inst. Channels	Permissive for closing Recirculating Pump Discharge Valve.
1	Reactor Low Pressure	50≤P≤75 psig	2 Inst. Channels	In conjunction with PCI signal permits closure of RHR (LPCI) injection valves.

TABLE 4.2.B

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor Water Level (7)	(1) (3)	Once/operating cycle	Once/day
2) **Drywell Pressure (7)	(1) (3)	Once/operating cycle	Once/day
*Drywell Pressure	(1)	Once/3 months	None
3) **Reactor Pressure (7)	(1) (3)	Once/operating cycle	Once/day
*Reactor Pressure	(1)	Once/3 months	None
4) Auto Sequencing Timers	NA	Once/operating cycle	None
5) ADS - LPCI or CS Pump Disch. Pressure Interlock	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	NA	None
7) Core Spray Sparger d/p	(1)	Once/6 months	Once/day
8) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
9) Steam Line High Temp. (HPCI & RCIC)	(1) (3)	Once/operating cycle	Once/day
10) Safeguards Area High Temp.	(1)	Once/3 months	None
11) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None
12) HPCI Suction Source Levels	(1)	Once/3 months	None
13) 4KV Emergency Power System Voltage Relays	Once/operating cycle	Once/5 year	None
14) ADS Relief Valves Bellows Pressure Switches	Once/operating cycle	Once/operating cycle	None
15) LPCI/Cross Connect Valve Position	Once/refueling cycle	N/A	N/A

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

Amendment No. 67

TABLE 4.2.E

MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Equipment Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
2) Floor Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
3) Air Sampling System	(1)	Once/3 months	Once/day

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) **Reactor Level	Once/operating cycle	Once Each Shift
*Reactor Level	Once/6 months	Once each shift
2) Reactor Pressure	Once/6 months	Once Each Shift
3) Drywell Pressure	Once/6 months	Once Each Shift
4) Drywell Temperature	Once/6 months	Once Each Shift
5) Suppression Chamber Temperature	Once/6 months	Once Each Shift
6) Suppression Chamber Water Level	Once/6 months	Once Each Shift
7) Control Rod Position	NA	Once Each Shift
8) Neutron Monitoring (APRM)	Twice Per Week	Once Each Shift

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 68 AND 67 TO FACILITY LICENSE NOS. DPR-33 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

1.0 Introduction

By letter dated August 27, 1979, and supplemented by letters dated November 5, 1979, January 30, 1980, February 13, 1980, and March 27, 1980, Philadelphia Electric Company (the licensee) requested amendments to Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3. The proposed amendments would revise the Technical Specifications to authorize the licensee to:

A. Replace existing pressure switches that sense drywell and reactor pressures with analog loops and

B. Modify two reactor water level indication loops to improve the reliability, accuracy, and response time of this instrumentation.

2.0 Discussion

2.1 Analog Loop Installation

The requested revision of the Technical Specification Tables 4.1.1, 4.1.2, 3.2.B, and 4.2.B reflects the planned replacement of existing pressure switches that sense reactor and drywell pressures with analog loops. Each loop will consist of a transmitter, indicator, and trip unit.

A detailed description of the proposed replacement is set forth in a document titled "Modification Description, Replacement of Pressure Switches with Analog Transmitters and Electronic Trip Units at Peach Bottom Atomic Power Station Units 2 and 3" which was attached to the licensee's application and incorporated therein by reference. The modification involves removing one device and substituting other devices to perform the same function. Changes in design bases, protective function, redundancy, trip point, and logic are not involved.

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2.2 The proposed instrument modifications are designed to:

1. Reduce undetected primary sensor element drift due to the utilization of a meter in each primary sensor signal loop;

2. Reduce the frequency of setpoint drift occurrences;

3. Provide indication for each primary sensor which will prove operability of the sensor;

4. Reduce the time Reactor Protection System logic must be in a half-scram condition to functionally test or calibrate a safety trip;

5. Reduce the functional test and calibration frequency for the primary sensor and to allow calibration of the primary sensor to be performed when the reactor is shut down for refueling.

6. Reduce likelihood of instrument valving errors;

7. Reduce the potential for instrument testing related scrams.

2.2 Modification of Reactor Level Indication Loops

The amendment to the Technical Specifications for Table 4.2.F is being requested to reflect the planned removal of existing reactor water level indication transmitters LITS 2-3-59A and B. They provide analog reactor water level signals to LI 2-3-85A and B and to LI 2-3-85AX and BX through a selector switch. The level transmitters will be removed and their function will be performed by existing channels of level transmitters LT 2-3-72A and B and the auxiliary analog output of trip units LISH 2-3-72A and B. Existing indicators will be replaced with new indicators that are compatible with the auxiliary analog output of the electronic trip units. The modification involves removing two level transmitters and substituting existing instruments to perform the same function. Changes in design basis, function, and redundancy are not involved.

The proposed instrument modifications are designed to:

1. Provide reliability, accuracy, and response time which are better than that of the existing instrumentation;

2. Provide the control room operators with reactor water level position indication from the same level transmitter that iniates the trip function;

3. Reduce likelihood of instrument valving errors.

3.0 Evaluation

The staff has reviewed previously the use of this type of equipment and found that, provided certain interface requirements were satisfied, this equipment is acceptable. Our letter of approval, dated June 27, 1978, is a part of General Electric Topical Report NEDO-21617-A dated December, 1978.

Although the licensee did not reference the General Electric Topical Report, the staff review did not result in the identification of any significant design differences between the equipment offered by General Electric and that used by the licensee. Furthermore, the staff noted that additional qualification testing, beyond that described in NEDO-21617-A, has been performed on the trip relays.

In order to satisfy some concerns about the validity of the reactor building accident response models (that would have to be resolved in order to demonstrate that the power supplies and inverters are qualified for the accident environment), the licensee is moving these units to the cable spreading room where the environment is known to be acceptable.

Based upon our review of the documentation, we conclude that the modifications proposed satisfy the constraints of our prior approval and are, therefore, acceptable. Based on the data submitted, we also concluded that:

1. The reliability, accuracy, and response time of the replacement instrumentation are better than that of the existing instrumentation.

2. Separation Criteria - the separation criteria of the original plant is unchanged. Separation is provided by locating equipment on separate racks and panels and by running cable in separated cable tray or conduit. The power supply used for an instrument channel is dependent on that channel's divisional assignment.

3. Single Failure Criterion - no new single failure events have been created, therefore, no single failure will result in any action not previously evaluated in the FSAR.

4. Qualification - all new equipment has been tested or analyzed to assure that the design environmental conditions and the design basis seismic requirements are met.

5. Testability - means are provided to test the trip units periodically by injecting a signal and observing the trip output. Operability of the analog loop is verified by periodic instrument checks.

4.0 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 the amendments.

5.0 Conclusion

We have concluded based on the considerations discussed above:

1. Because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, 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: May 5, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET 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 Amendment No. 68 and 67 to Facility Operating License Nos. DPR-44 and DPR-56, issued to the Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications for operation of the Peach Bottom Atomic Power Station, 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 Technical Specifications to authorize Philadelphia Electric Company to replace existing pressure switches that sense drywell and reactor pressure with analog loops and to modify two reactor water level indication loops to improve the reliability, accuracy, and response time of this instrumentation.

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

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§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 amendment dated August 27, 1979, as supplemented by letters dated November 5, 1979, January 30, 1980, February 13, 1980, and March 27, 1980, (2) Amendment Nos. 68 and 67 to License Nos. DPR-44 and DPR-56, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 "H" Street, N.W., Washington, D. C. and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director: Division of Operating Reactors.

Dated at Bethesda, Maryland this 5th day of May 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

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