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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 30 and 29 to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units Nos. 2 and 3. These amendments consist of changes to the Technical Specifications and are in response to your request dated November 22, 1976.

These amendments will change the Technical Specification to reflect the following planned modifications to the Peach Bottom facility: (1) replacement of existing pressure and differential pressure switches which sense condenser vacuum, reactor water level and main steam line flow, with analog loops, and (2) the addition of an automatic isolation signal to the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) System turbine-exhaust vacuum-breaker isolation valves.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

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

Enclosures:
1. Amendments Nos. 30 and 29
2. Safety Evaluation
3. Federal Register Notice

cont'd

cc:
See next page

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DATE	12/22/76	12/22/76	12/3/76	1/3/77	

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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. 30
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, (the licensees) dated November 22, 1976, 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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of the completion of modifications to the pressure sensing devices and the vacuum breaker isolation valves as described in the licensee's application dated November 22, 1976.

George Lear

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: January 3, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 30

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace pages 41-44, 51-53, 79-81, 87, 88, 181-183, 187, 188, 199-202 with the attached revised pages. No change has been made on pages 51, 79, 88, 182, 188, 199, and 202. Insert new page 53a.

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	B 1	Trip Output Relays (4)	Once/week.
Inoperative	B 1	Trip Output Relays (4)	Once/week.
Downscale	B 1	Trip Output Relays (4)	Once/week.
Flow Bias	B 1	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)	B 2	Trip Channel and Alarm (4)	Every 1 month (1).
High Drywell Pressure	A	Trip Channel and Alarm	Every 1 month (1).
Reactor Low Water Level (5) (6)	B 2	Trip Channel and Alarm (4)	Every 1 month (1).

Amendment No. 30

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)	B 2	Trip Channel and Alarm (4)	Every 1 month
Main Steam Line High Radiation	B 1	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	A	Trip Channel and Alarm	Every 3 months

NOTES FOR TABLE 4.1.1

1. Initially once every month. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of PBAPS. The failure rate data must be reviewed and approved by the NRC prior to any change in the once-a-month frequency.

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

3. Functional tests 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.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the functional test program.

6. These channels consist of analog transmitters, indicators and electronic trip units. Instrument checks shall be performed once per day.

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 Shutdowns	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	Every 3 months.
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 cycl(
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.

4.1 BASES

A. The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (6). This concept was specifically adapted to the one out of two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failures such as blown fuses, ruptured bourdon tubes, faulted amplifiers, and faulted cables, which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups for functional testing. These are:

A. On-off sensors that provide a scram trip function.

B. Analog devices coupled with bi-stable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed at shutdown.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.99999 probability of success (at the 50% confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval is planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an a adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95% confidence level is proposed. With the (1 out of 2) X (2) logic, this

4.1 BASES (Cont'd.)

requires that each sensor have an availability of 0.993 at the 95% confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history (6). To facilitate the implementation of this technique, Figure 4.1.1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.

2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T (M= nT).

3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1.1.

4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.

5. A test interval of 1 month will be used initially until a trend is established, which is based on system availability analysis and good engineering judgment plus operating experience.

Group (B1)* devices utilize an analog sensor followed by an amplifier and a bistable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" mid-scale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purpose of analysis, it is assumed that this rare failure will be detected within two hours.

The bi-stable trip circuit which is a part of the Group (B1) devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

---

(6) Reliability of Engineered Safety Features as a Function of Testing Frequency, I.M. Jacobs, "Nuclear Safety", Vol. 9, No. 4, July-Aug. 1968, pp. 310-312.

* See note following Group (B2)

4.1 BASES (Cont'd)

A study was conducted of the instrumentation channels included in the Group (B1) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20 X 10-6 failure/hour. The bi-stable trip circuits are predicted to have unsafe failure rate of less than 2 X 10^{-6} failures/hour. Considering the two hour monitoring interval for the analog devices as assumed above and a weekly test interval for the bi-stable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bi-stable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1.1. There are numerous identical bi-stable devices used throughout the plant's instrumentation system. Therefore, significant data on the failure rates for the bi-stable devices should be accumulated rapidly.

The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. The flow biasing network is functionally tested at least once per month and in addition, cross calibration checks of the flow input to the flow biasing network can be made during the functional test by direct meter reading. There are several instruments which must be calibrated and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (B2)* devices utilize an analog sensor followed by an amplifier and a bistable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" mid-scale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the

* See note following Group (B2)

other three. For purpose of analysis, it is assumed that this rare failure will be detected within twenty-four hours.

The bi-stable trip circuit which is a part of the Group (B2) devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in Group (B2) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 2 X 10^{-5} failures/hour. The bistable trip circuits are predicted to have an unsafe failure rate of less than 9 X 10^{-6} failures/hour. Considering the twenty-four hour monitoring interval for the analog devices and a monthly test interval for the bi-stable trip circuits, the design reliability goal of 0.993 per channel is attained. As described in the above discussion for Group (A) devices, a per channel reliability of 0.993 yields an overall reliability of 0.9999 for this instrumentation.

Note: Analog Loop indicators for Group (B1) are located in the Control Room and therefore can be checked once per shift. Analog Loop indicators for Group (B2)are located in the plant adjacent to the applicable equipment and therefore can be checked once per day.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semi-conductor devices and detectors that drift or lose sensitivity.

TABLE 3.2.G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum Number of Operable Instrument Channels per Trip (System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
1	Reactor High Pressure	$\leq$1120 psig	4	(2)
1	Reactor Low Water Level	$\geq$ -49 in. indicated level	4	(2)

Notes for Table 3.2.G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for each operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.

TABLE 4.2.A

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

Instrument Channel (5)	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor High Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2) Reactor Low-Low Water Level (7)	(1)(3)	Once/operating cycle	Once/day
3) Main Steam High Temp.	(1) (3)	Once/operating cycle	Once/day
4) Main Steam High Flow (7)	(1)(3)	Once/operating cycle	Once/day
5) Main Steam Low Pressure	(1)	Once/3 months	None
6) Reactor Water Cleanup High Flow	(1)	Once/3 months	Once/day
7) Reactor Water Cleanup High Temp.	(1)	Once/3 months	None

Logic System Functional Test (4) (6)	Frequency
1) Main Steam Line Isolation Vvs. Main Steam Line Drain Vvs. Reactor Water Sample Vvs.	Once/6 months
2) RHR - Isolation Vv. Control Shutdown Cooling Vvs. Head Spray	Once/6 months
3) Reactor Water Cleanup Isolation	Once/6 months
4) Drywell Isolation Vvs. TIP Withdrawal Atmospheric Control Vvs. Sump Drain Valves	Once/6 months
5) Standby Gas Treatment System Reactor Building Isolation	Once/6 months

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	(1)	Once/3 months	None
3)	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 outage	N/A	N/A

NOTES FOR TABLES 4.2.A THROUGH 4.2.F

1. Initially once every month. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of PBAPS. The failure rate data must be reviewed and approved by the AEC prior to any change in the once-a-month frequency.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be peformed within 24 hours before each startup or controlled shutdown with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.
These instrument channels will be calibrated using simulated electrical signals.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2.A since they are tested on Table 4.1.2.

6. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

7. These channels consist of analog transmitters, indicators and electronic Trip units.

TABLE 4.2.G

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RECIRCULATION PUMP TRIP

Instrument Channel	Instrument Functional Check	Calibration Frequency
Reactor High Pressure	Once/refueling cycle	Once/refueling cycle
Reactor Low Water Level	Once/refueling cycle	Once/refueling cycle

Logic System Functional Test	Frequency
Recirculation Pump Trip	Once/refueling cycle

TABLE 3.7.1 (cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
2D	Drywell equipment drain discharge isolation valves		2	5	0	GC
2D	Drywell floor drain discharge isolation valves		2	5	0	GC
2D	Traveling in-core probe		5	NA	C	SC
4A	HPCI steam line drains		2	NA	0	GC
5A	RCIC steam line drains		2	NA	0	GC
5A	RCIC condensate pump drain		2	NA	0	GC
4A	HPCI condensate pump drain		2	NA	C	SC
2D	Torus water filter pumps suction isolation valves		2	NA	0	GC
4B	HPCI Turbine Exhaust Vacuum Breaker Isolation Valve	1		15	0	GC
5B	RCIC Turbine Exhaust Vacuum Breaker Isolation Valve	1		15	0	GC

NOTES FOR TABLE NO. 3.7.1

Key: 0 = Open

C = Closed

SC = Stays Closed

GC = Goes Closed

Note: Isolation groupings are as follows:

GROUP 1 : The valves in Group 1 are actuated by any one of the following conditions:

1. Reactor vessel low-low water level.
2. Main steam line high radiation.
3. Main steam line high flow.
4. Main steam line space high temperature.
5. Main steam line low pressure (RUN mode only).

GROUP 2A: The valves in Group 2A are actuated by any one of the following conditions:

1. Reactor vessel low water level.
2. Reactor water cleanup system heat exchanger discharge high temperature.
3. Reactor water cleanup system suction line break.
4. Standby liquid control system actuation.

GROUP 2B: The valves in Group 2B are actuated by any one of the following conditions:

1. Reactor vessel low water level.
2. High drywell pressure.
3. Reactor high pressure of shutdown mode.

GROUP 2C: The valves in Group 2C are actuated by any one of the following conditions:

1. Reactor low water level.
2. High reactor vessel pressure, (600 PSIG)
3. High drywell pressure.

GROUP 2D: The valves in Group 2D are actuated by the following conditions:

1. High drywell pressure.
2. Reactor low water level.

GROUP 3 : The valves in Group 3 are actuated by any one of the following conditions:

1. Reactor vessel low water level.

2. High drywell pressure.

3. Reactor building ventilation exhaust high radiation.

4. Refuel floor ventilation exhaust high radiation.

GROUP 4 : The valves in Group are actuated by any one of the following conditions:

1. HPCI steam line high flow.

2. HPCI steam line space high temperature.

3. HPCI steam line low pressure.

GROUP 4A: The valves in Group 4A are actuated by either of the following conditions:

1. Reactor vessel low-low water level.

2. High drywell pressure.

GROUP 4B: The valve in Group 4B is actuated when both of the following conditions are present:

1. High drywell pressure.

2. HPCI steam line low pressure.

GROUP 5 : The valves in Group 5 are actuated by any one of the following conditions:

1. RCIC steam line high flow.

2. RCIC steam line space high temperature.

3. RCIC steam line low pressure.

GROUP 5A: The valves in Group 5A are actuated by the following condition:

1. Reactor vessel low-low water level.

GROUP 5B: The valve in Group 5B is actuated when both of the following conditions are present:

1. High drywell pressure.

2. RCIC steam line low pressure.

TABLE 3.7. (Cont'd.)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

Pen No.		Notes
211A	MO-10-38B; MO-10-39B; MO-10-34B	(1)(2)(4)(5)(9)
211A	SV-4951B; SV-4950B	(1)(2)(4)(5)
211B	MO-10-38A; MO-10-39A; MO-10-34B	(1)(2)(4)(5)(9)
211B	SV-4951A; SV-4950A	(1)(2)(4)(5)
212	Check Valve 13-50; AO-4240; AO-4241	(1)(2)(4)(5)(9)
214	Check Valve 23-65; AO-4247; AO-4248	"
217B	MO-4244; MO-4244A	(1) (2) (4) (5) (9)
218A	AO-2968	(1)(2)(4)(5)(10)
218B	SV-2671A; SV-2978A	(1)(2)(4)(5)
219	AO-2511; AO-2512; AO-2513; AO-2514	(1)(2)(4)(5)(9)
219	SV-2671F; SV-2978F; SV-4960A; SV-4961A; SV-4966A	(1)(2)(4)(5)
221	Check Valve 13-38	(1)(2)(4)(5)(9)
223	Check Valve 23-56	"
225	MO-13-40; MO-13-39	"
225	MO-13-70; MO-14-71	"
227	MO-23-58; MO 23-57	"

NOTES FOR TABLES 3.7.2 THROUGH 3.7.4

(1) Minimum test duration for all valves and penetrations listed is one hour.

(2) Test pressures of at least 49.1 psig for all valves and penetrations except MSIV's which are tested at 25 psig.

(3) MSIV's acceptable leakage is 11.5 scfh/valve of air.

(4) The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.60 La.

(5) Local leak tests on all testable isolation valves shall be performed each operating cycle but in no case at intervals greater than 2 years.

(6) Local leak tests on all testable penetrations shall be performed each operating cycle but in no case at intervals greater than 2 years.

(7) Air locks shall be tested at 6-month intervals except if air locks are not opened during this interval, in which case tests shall be performed after each opening but no interval may be longer than one year.

(8) The personnel air locks are tested at 10 psig.

(9) Identifies isolation valves that are tested by applying pressure between the inboard and outboard valves. Inboard valve is tested in conservative direction.

(10) Identifies outboard valve tested in conservative direction.

3.7.D & 4.7.D BASES

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

Group 1: Actuation for valves associated with the isolation of the main steam system. The main steam lines are isolated by reactor vessel low-low water level in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature.

Group 2: Actuation for valves associated with the isolation of the reactor auxiliary systems. Some of the reactor auxiliary systems such as the RWCU and RHR shutdown cooling systems connect into the reactor coolant boundary while others such as the drywell equipment and floor drain discharge valves do not penetrate the reactor coolant boundary. Group 2 actuation is subdivided as follows:

Group 2A - process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature at the cleanup system heat exchanger/outlet or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low level isolation is provided. An alarm of high temperature in the cleanup system area will provide an indication of suction line break resulting in manual isolation of the system. During actuation of the standby liquid control system, the cleanup system is isolated.

Group 2B - isolation valves are not normally in use and are closed by reactor vessel low water level, high drywell pressure or high reactor pressure of the shutdown mode.

Group 2C - isolation valves can only be opened when the reactor is at low pressure and the core standby cooling systems are not required. Also, since the reactor vessel could potentially be drained through these process lines, these valves are closed by low water level.

Group 2D - line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure indicates a possible accident condition.

Group 3: Actuation for isolation of valves and dampers associated with the ventilation systems. Group 3 lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation, or refueling floor ventilation high radiation which would indicate a possible refueling accident. The group 3 isolation signals also "isolate" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the group 3 isolation signal by a transient or spurious signal.

Groups 4 and 5: Actuation associated with process lines that are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Group 4 and 5 process lines are therefore indicative of a condition which would render them inoperable. Groups 4 and 5 are subdivided as follows:

Group 4A - process lines are closed on reactor low water level (490") or high drywell pressure. These close on the same signal that initiates HPCIS to ensure that the valves are not open when HPCIS action is required.

Group 4B - line is isolated on high drywell pressure if the HPCI System has been rendered inoperable by low steam line pressure.

Group 5A - process lines are closed only on reactor low water level (490"). These close on the same signal that initiates RCICS to ensure that the valves are not open when RCICS action is required.

Group 5B - line is isolated on high drywell pressure if the RCIC system has been rendered inoperable by low steam line pressure.

The maximum closure times for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

These valves are highly reliable, have a low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment. A program for periodic testing and examination of the excess flow check valves is performed as follows:

1. Vessel at pressure sufficient to actuate valves. This could be at time of vessel hydro following a refueling outage.

2. Isolate sensing line from its instrument at the instrument manifold.

3. Provide means for observing and collecting the instrument drain or vent valve flow.

4. Open vent or drain valve.

    a) Observe flow cessation and any leakage rate.

    b) Observe closed position light actuation.

    c) Reset valve after test completion.

5. The head seal leak detection line cannot be tested in this manner. This valve will not be exposed to primary system pressure except under unlikely conditions of seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source and therefore this valve need not be tested. This valve is in a sensing line that is not safety related.

6. Valves will be accepted if a marked decrease in flow rate is observed, the leakage rate is acceptable, and position closure light is observed.

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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. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, (the licensees) dated November 22, 1976, 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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of the completion of modifications to the pressure sensing devices and the vacuum breaker isolation valves as described in the licensee's application dated November 22, 1976.

George Lear

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: January 3, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 29

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace pages 41-44, 51-53, 79-81, 87, 88, 181-183, 187, 188, 199-202 with the attached revised pages. No change has been made on pages 51, 79, 88, 182, 188, 199, and 202. Insert new page 53a.

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	B 1	Trip Output Relays (4)	Once/week.
Inoperative	B 1	Trip Output Relays (4)	Once/week.
Downscale	B 1	Trip Output Relays (4)	Once/week.
Flow Bias	B 1	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)	B 2	Trip Channel and Alarm (4)	Every 1 month (1).
High Drywell Pressure	A	Trip Channel and Alarm	Every 1 month (1).
Reactor Low Water Level (5) (6)	B 2	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)	B 2	Trip Channel and Alarm (4)	Every 1 month (1)
Main Steam Line High Radiation	B 1	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	A	Trip Channel and Alarm	Every 3 months

NOTES FOR TABLE 4.1.1

1. Initially once every month. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of PBAPS. The failure rate data must be reviewed and approved by the NRC prior to any change in the once-a-month frequency.

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

3. Functional tests 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.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the functional test program.

6. These channels consist of analog transmitters, indicators and electronic trip units. Instrument checks shall be performed once per day.

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 Shutdowns	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	Every 3 months.
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 cycl(
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.

4.1 BASES

A. The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (6). This concept was specifically adapted to the one out of two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failures such as blown fuses, ruptured bourdon tubes, faulted amplifiers, and faulted cables, which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups for functional testing. These are:

A. On-off sensors that provide a scram trip function.

B. Analog devices coupled with bi-stable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed at shutdown.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.99999 probability of success (at the 50% confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval is planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an a adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95% confidence level is proposed. With the (1 out of 2) X (2) logic, this

4.1 BASES (Cont'd.)

requires that each sensor have an availability of 0.993 at the 95% confidence level. This level of availibility may be maintained by adjusting the test interval as a function of the observed failure history (6). To facilitate the implementation of this technique, Figure 4.1.1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.

2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T (M= nT).

3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1.1.

4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.

5. A test interval of 1 month will be used initially until a trend is established, which is based on system availability analysis and good engineering judgment plus operating experience.

Group (B1)* devices utilize an analog sensor followed by an amplifier and a bistable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" mid-scale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purpose of analysis, it is assumed that this rare failure will be detected within two hours.

The bi-stable trip circuit which is a part of the Group (B1) devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

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(6) Reliability of Engineered Safety Features as a Function of Testing Frequency, I.M. Jacobs, "Nuclear Safety", Vol. 9, No. 4, July-Aug. 1968, pp. 310-312.

* See note following Group (B2)

4.1 BASES (Cont'd)

A study was conducted of the instrumentation channels included in the Group (B1) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20 X 10-6 failure/hour. The bi-stable trip circuits are predicted to have unsafe failure rate of less than 2 X 10^{-6} failures/hour. Considering the two hour monitoring interval for the analog devices as assumed above and a weekly test interval for the bi-stable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bi-stable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1.1. There are numerous identical bi-stable devices used throughout the plant's instrumentation system. Therefore, significant data on the failure rates for the bi-stable devices should be accumulated rapidly.

The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. The flow biasing network is functionally tested at least once per month and in addition, cross calibration checks of the flow input to the flow biasing network can be made during the functional test by direct meter reading. There are several instruments which must be calibrated and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (B2)* devices utilize an analog sensor followed by an amplifier and a bistable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" mid-scale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the

* See note following Group (B2)

other three. For purpose of analysis, it is assumed that this rare failure will be detected within twenty-four hours.

The bi-stable trip circuit which is a part of the Group (B2) devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in Group (B2) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 2×10^{-5} failures/hour. The bistable trip circuits are predicted to have an unsafe failure rate of less than 9×10^{-6} failures/hour. Considering the twenty-four hour monitoring interval for the analog devices and a monthly test interval for the bi-stable trip circuits, the design reliability goal of 0.993 per channel is attained. As described in the above discussion for Group (A) devices, a per channel reliability of 0.993 yields an overall reliability of 0.9999 for this instrumentation.

Note: Analog Loop indicators for Group (B1) are located in the Control Room and therefore can be checked once per shift. Analog Loop indicators for Group (B2) are located in the plant adjacent to the applicable equipment and therefore can be checked once per day.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semi-conductor devices and detectors that drift or lose sensitivity.

TABLE 3.2.G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum Number of Operable Instrument Channels per Trip (System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
1	Reactor High Pressure	$\leq$1120 psig	4	(2)
1	Reactor Low Water Level	$\geq$ -49 in. indicated level	4	(2)

Notes for Table 3.2.G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for each operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.

TABLE 4.2.A

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

	Instrument Channel (5)	Instrument Functional Test	Calibration Frequency	Instrument Check
1)	Reactor High Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2)	Reactor Low-Low Water Level (7)	(1) (3)	Once/operating cycle	Once/day
3)	Main Steam High Temp.	(1) (3)	Once/operating cycle	Once/day
4)	Main Steam High Flow (7)	(1) (3)	Once/operating cycle	Once/day
5)	Main Steam Low Pressure	(1)	Once/3 months	None
6)	Reactor Water Cleanup High Flow	(1)	Once/3 months	Once/day
7)	Reactor Water Cleanup High Temp.	(1)	Once/3 months	None

	Logic System Functional Test (4) (6)	Frequency
1)	Main Steam Line Isolation Vvs. Main Steam Line Drain Vvs. Reactor Water Sample Vvs.	Once/6 months
2)	RHR - Isolation Vv. Control Shutdown Cooling Vvs. Head Spray	Once/6 months
3)	Reactor Water Cleanup Isolation	Once/6 months
4)	Drywell Isolation Vvs. TIP Withdrawal Atmospheric Control Vvs. Sump Drain Valves	Once/6 months
5)	Standby Gas Treatment System Reactor Building Isolation	Once/6 months

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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	(1)	Once/3 months	None
3)	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 outage	N/A	N/A

NOTES FOR TABLES 4.2.A THROUGH 4.2.F

1. Initially once every month. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of PBAPS. The failure rate data must be reviewed and approved by the AEC prior to any change in the once-a-month frequency.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be peformed within 24 hours before each startup or controlled shutdown with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.
These instrument channels will be calibrated using simulated electrical signals.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2.A since they are tested on Table 4.1.2.

6. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

7. These channels consist of analog transmitters, indicators and electronic Trip units.

TABLE 4.2.G

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RECIRCULATION PUMP TRIP

Instrument Channel	Instrument Functional Check	Calibration Frequency
Reactor High Pressure	Once/refueling cycle	Once/refueling cycle
Reactor Low Water Level	Once/refueling cycle	Once/refueling cycle

Logic System Functional Test	Frequency
Recirculation Pump Trip	Once/refueling cycle

TABLE 3.7.1 (cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
2D	Drywell equipment drain discharge isolation valves		2	5	0	GC
2D	Drywell floor drain discharge isolation valves		2	5	0	GC
2D	Traveling in-core probe		5	NA	C	SC
4A	HPCI steam line drains		2	NA	0	GC
5A	RCIC steam line drains		2	NA	0	GC
5A	RCIC condensate pump drain		2	NA	0	GC
4A	HPCI condensate pump drain		2	NA	C	SC
2D	Torus water filter pumps suction isolation valves		2	NA	0	GC
4B	HPCI Turbine Exhaust Vacuum Breaker Isolation Valve	1		15	0	GC
5B	RCIC Turbine Exhaust Vacuum Breaker Isolation Valve	1		15	0	GC

NOTES FOR TABLE NO. 3.7.1

Key:
- 0 = Open
- C = Closed
- SC = Stays Closed
- GC = Goes Closed

Note: Isolation groupings are as follows:

GROUP 1 : The valves in Group 1 are actuated by any one of the following conditions:

1. Reactor vessel low-low water level.
2. Main steam line high radiation.
3. Main steam line high flow.
4. Main steam line space high temperature.
5. Main steam line low pressure (RUN mode only).

GROUP 2A: The valves in Group 2A are actuated by any one of the following conditions:

1. Reactor vessel low water level.
2. Reactor water cleanup system heat exchanger discharge high temperature.
3. Reactor water cleanup system suction line break.
4. Standby liquid control system actuation.

GROUP 2B: The valves in Group 2B are actuated by any one of the following conditions:

1. Reactor vessel low water level.
2. High drywell pressure.
3. Reactor high pressure of shutdown mode.

GROUP 2C: The valves in Group 2C are actuated by any one of the following conditions:

1. Reactor low water level.
2. High reactor vessel pressure, (600 PSIG)
3. High drywell pressure.

GROUP 2D: The valves in Group 2D are actuated by the following conditions:

1. High drywell pressure.
2. Reactor low water level.

GROUP 3 : The valves in Group 3 are actuated by any one of the following conditions:

1. Reactor vessel low water level.

2. High drywell pressure.

3. Reactor building ventilation exhaust high radiation.

4. Refuel floor ventilation exhaust high radiation.

GROUP 4 : The valves in Group are actuated by any one of the following conditions:

1. HPCI steam line high flow.

2. HPCI steam line space high temperature.

3. HPCI steam line low pressure.

GROUP 4A: The valves in Group 4A are actuated by either of the following conditions:

1. Reactor vessel low-low water level.

2. High drywell pressure.

GROUP 4B: The valve in Group 4B is actuated when both of the following conditions are present:

1. High drywell pressure.

2. HPCI steam line low pressure.

GROUP 5 : The valves in Group 5 are actuated by any one of the following conditions:

1. RCIC steam line high flow.

2. RCIC steam line space high temperature.

3. RCIC steam line low pressure.

GROUP 5A: The valves in Group 5A are actuated by the following condition:

1. Reactor vessel low-low water level.

GROUP 5B: The valve in Group 5B is actuated when both of the following conditions are present:

1. High drywell pressure.

2. RCIC steam line low pressure.

TABLE 3.7. (Cont'd.)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

Pen No.		Notes
211A	MO-10-38B; MO-10-39B; MO-10-34B	(1)(2)(4)(5)(9)
211A	SV-4951B; SV-4950B	(1)(2)(4)(5)
211B	MO-10-38A; MO-10-39A; MO-10-34B	(1)(2)(4)(5)(9)
211B	SV-4951A; SV-4950A	(1)(2)(4)(5)
212	Check Valve 13-50; AO-4240; AO-4241	(1)(2)(4)(5)(9)
214	Check Valve 23-65; AO-4247; AO-4248	"
217B	MO-4244; MO-4244A	(1) (2) (4) (5) (9)
218A	AO-2968	(1)(2)(4)(5)(10)
218B	SV-2671A; SV-2978A	(1)(2)(4)(5)
219	AO-2511; AO-2512; AO-2513; AO-2514	(1)(2)(4)(5)(9)
219	SV-2671F; SV-2978F; SV-4960A; SV-4961A; SV-4966A	(1)(2)(4)(5)
221	Check Valve 13-38	(1)(2)(4)(5)(9)
223	Check Valve 23-56	"
225	MO-13-40; MO-13-39	"
225	MO-13-70; MO-14-71	"
227	MO-23-58; MO-23-57	"

Amendment No. 29

NOTES FOR TABLES 3.7.2 THROUGH 3.7.4

(1) Minimum test duration for all valves and penetrations listed is one hour.

(2) Test pressures of at least 49.1 psig for all valves and penetrations except MSIV's which are tested at 25 psig.

(3) MSIV's acceptable leakage is 11.5 scfh/valve of air.

(4) The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.60 La.

(5) Local leak tests on all testable isolation valves shall be performed each operating cycle but in no case at intervals greater than 2 years.

(6) Local leak tests on all testable penetrations shall be performed each operating cycle but in no case at intervals greater than 2 years.

(7) Air locks shall be tested at 6-month intervals except if air locks are not opened during this interval, in which case tests shall be performed after each opening but no interval may be longer than one year.

(8) The personnel air locks are tested at 10 psig.

(9) Identifies isolation valves that are tested by applying pressure between the inboard and outboard valves. Inboard valve is tested in conservative direction.

(10) Identifies outboard valve tested in conservative direction.

3.7.D & 4.7.D BASES

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

Group 1: Actuation for valves associated with the isolation of the main steam system. The main steam lines are isolated by reactor vessel low-low water level in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature.

Group 2: Actuation for valves associated with the isolation of the reactor auxiliary systems. Some of the reactor auxiliary systems such as the RWCU and RHR shutdown cooling systems connect into the reactor coolant boundary while others such as the drywell equipment and floor drain discharge valves do not penetrate the reactor coolant boundary. Group 2 actuation is subdivided as follows:

Group 2A - process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature at the cleanup system heat exchanger/outlet or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low level isolation is provided. An alarm of high temperature in the cleanup system area will provide an indication of suction line break resulting in manual isolation of the system. During actuation of the standby liquid control system, the cleanup system is isolated.

Group 2B - isolation valves are not normally in use and are closed by reactor vessel low water level, high drywell pressure or high reactor pressure of the shutdown mode.

Group 2C - isolation valves can only be opened when the reactor is at low pressure and the core standby cooling systems are not required. Also, since the reactor vessel could potentially be drained through these process lines, these valves are closed by low water level.

Group 2D - line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure indicates a possible accident condition.

Group 3: Actuation for isolation of valves and dampers associated with the ventilation systems. Group 3 lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation, or refueling floor ventilation high radiation which would indicate a possible refueling accident. The group 3 isolation signals also "isolate" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the group 3 isolation signal by a transient or spurious signal.

Groups 4 and 5: Actuation associated with process lines that are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Group 4 and 5 process lines are therefore indicative of a condition which would render them inoperable. Groups 4 and 5 are subdivided as follows:

Group 4A - process lines are closed on reactor low water level (490") or high drywell pressure. These close on the same signal that initiates HPCIS to ensure that the valves are not open when HPCIS action is required.

Group 4B - line is isolated on high drywell pressure if the HPCI System has been rendered inoperable by low steam line pressure.

Group 5A - process lines are closed only on reactor low water level (490"). These close on the same signal that initiates RCICS to ensure that the valves are not open when RCICS action is required.

Group 5B - line is isolated on high drywell pressure if the RCIC system has been rendered inoperable by low steam line pressure.

The maximum closure times for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

These valves are highly reliable, have a low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment. A program for periodic testing and examination of the excess flow check valves is performed as follows:

1. Vessel at pressure sufficient to actuate valves. This could be at time of vessel hydro following a refueling outage.

2. Isolate sensing line from its instrument at the instrument manifold.

3. Provide means for observing and collecting the instrument drain or vent valve flow.

4. Open vent or drain valve.

   a) Observe flow cessation and any leakage rate.

   b) Observe closed position light actuation.

   c) Reset valve after test completion.

5. The head seal leak detection line cannot be tested in this manner. This valve will not be exposed to primary system pressure except under unlikely conditions of seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source and therefore this valve need not be tested. This valve is in a sensing line that is not safety related.

6. Valves will be accepted if a marked decrease in flow rate is observed, the leakage rate is acceptable, and position closure light is observed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 30 TO FACILITY LICENSE NO. DPR-44 AND

AMENDMENT NO. 29 TO FACILITY LICENSE NO. DPR-56

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

Introduction

By letter dated November 22, 1976, Philadelphia Electric Company (PECO) requested amendments to Facility Operating Licenses Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station, Units Nos. 2 and 3. The proposed amendments would change the Technical Specifications to reflect the following modifications to the Peach Bottom facility: (1) replacement of existing pressure and differential pressure switches which sense condenser vacuum, reactor water level and main steam line flow with analog loops, and (2) the addition of an automatic isolation signal to the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) Systems turbine-exhaust vacuum-breaker isolation valves.

Discussion

Selected differential-pressure and pressure switches used for flow, level and pressure measurement are to be replaced with analog transmitters and electronic trip units to increase the reliability of these sensing devices and to reduce the manpower requirements for functional testing and calibration of instruments. The licensee's proposed changes affect only the method of generating the trip signals. All trip functions would remain the same with no changes in design basis, protective function, redundancy, trip point setting or logic.

The automatic isolation signal to be added to the RCIC and HPCI turbine-exhaust vacuum-breaker isolation valves consists of a high drywell pressure signal and a low reactor pressure signal combined in an "AND" logic. The high drywell pressure signal indicates a need for containment isolation. The low reactor pressure signal is a permissive that allows isolation only after the reactor pressure has dropped to a value that renders the RCIC or HPCI system inoperable. The automatic isolation feature is being installed on these valves to conform the HPCI and RCIC systems to Criterion 56, Appendix A to 10 CFR Part 50 which addresses Primary Containment isolation.

Evaluation

Replacement of Existing Pressure Sensing Devices

Since the modification involves removing one device and substituting other devices to perform the same function, the evaluation can be limited to demonstrating that the replacement devices meet the reliability, accuracy and response time requirements of the instrument channels. The modification involves no changes in design basis, protective function, redundancy, trip point, logic, or the Commission's requirements which were applicable to the original design. The design reliability goal of the existing device is 0.993. Our analysis of the replacement instrument channels verifies that the product of the individual reliabilities of the transmitter and the trip unit yields a channel reliability greater than 0.993. Thus, the replacement channel satisfies the design reliability goal of the existing instrumentation. In all cases the licensee has verified that the response time and accuracy requirements of the instrument channels are met by the replacement devices. Based upon the foregoing we conclude that these proposed modifications are acceptable.

Automatic Isolation Signal for RCIC and HPCI Turbine-Exhaust Vacuum Breakers

As previously discussed, this change is being proposed to bring these systems into conformance with Criterion 56, Appendix A to 10 CFR Part 50. We have reviewed the licensee's proposed change and conclude that the addition of an automatic isolation signal to the control logic of these valves does not have any adverse impact on safety because: (1) the automatic isolation function will not preclude normal remote operation of the associated valve and (2) inadvertent closure of these valves, through spurious isolation signals, is minimized by combining the sensor inputs in a one-out-of-two taken twice logic. Based upon the foregoing, we conclude that these proposed modifications are also acceptable.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 3, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

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

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 30 and 29 to Facility Operating Licenses Nos. DPR-44 and DPR-56, respectively, issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications for operation of the Peach Bottom Atomic Power Station, Units Nos. 2 and 3, located in Peach Bottom, York County, Pennsylvania. The amendments are effective as of the date of issuance.

The amendments will change the Technical Specifications to reflect the following planned modifications to the Peach Bottom facility: (1) replacement of existing pressure and differential pressure switches which sense condenser vacuum, reactor water level and main steam line flow, with analog loops, and (2) the addition of an automatic isolation signal to the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injections (HPCI) Systems turbine-exhaust vacuum-breaker isolation valves.

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 November 22, 1976, (2) Amendments Nos. 30 and 29 to Licenses 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 Martin Memorial Library, 159 E. Market Street, York, Pennsylvania 17401.

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 3 day of January 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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