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Docket Nos. 50-277
and 50-278

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

Dear Mr. Bauer:

The Commission has issued the enclosed Amendments Nos. 47 and 47 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 requests dated May 26 and September 5, 1978.

The amendments revise the Technical Specifications which relate to: (1) the instrumentation that initiates or controls the core and containment cooling systems, (2) Administrative Controls, (3) addition of safety related snubbers, (4) certain revisions that would conform to the staff's Standard Technical Specifications, and (5) various editorial changes to clarify the meaning or correct errors in the existing specifications.

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

Sincerely,

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

Const I

Enclosures:
1. Amendment No. 47 to DPR-44
2. Amendment No. 40 to DPR-56
3. Safety Evaluation
4. Notice

cc w/enclosures:
see next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE➤	ORB#3	ORB#3	OELD	ORB#3
SURNAME➤	*SSheppard	*DVerrelli:ar	*MCutchin	TIppolito
DATE➤	9/28/78	9/28/78	10/4/78	10/ /78

NRC FORM 318 (9-76) NRCM 0240

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Docket Nos. 50-277
and 50-278

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

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Sincerely,

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

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

Subject to changes noted in SER

cc w/enclosures:
See next page

OFFICE➤		ORB#3	ORB#3	OELD	ORB#3	
SURNAME➤		SSheppard	DVerrelli:ar	CUTCHIN	TIppolito	
DATE➤		9/ 28 /78	9/ 28 /78	10/ 4 /78	10/ 10 /78	

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Philadelphia Electric Company

cc:

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

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. DPR-44

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

   A. The applications for amendment by Philadelphia Electric Company, et al, (the licensee) dated May 26 and September 5, 1978, comply 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 applications, 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. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 10, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Revise Appendix A Technical Specifications as follows:

Remove	Replace
iii	iii
67	67
68*	68*
102a	102a
115*	115*
116	116
125	125
126	126
127	127
177*	177*
178	178
249	249
250	250
253	253
254	254
254b	254b
257	257
262	262
263	deleted
264	deleted

*No changes on this page

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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	Core Spray Pump Start Timer	6 ± 1 sec 10 ± 1 sec	4 timers 4 timers	In conjunction with loss of power initiates the starting of CSCS pumps.
1	LPCI Pump Start Timer	$0 < t < 1$ sec 5 ± 1 sec	2 timers 2 timers	This Specification shall be effective until replacement of the "$0 < t < 1$ sec" timers with auxiliary relays.
2	LPCI Pump Start Timer (Two pumps)	5 ± 1 sec	4 timers	This Specification shall be effective after replacement of the "$0 < t < 1$ sec" timers with auxiliary relays.
1	Auto Blowdown Timer	$90 \leq t \leq 120$	2 timers	In conjunction with Low Reactor Water Level, High Drywell Pressure and LPCI or Core Spray Pump running interlock, initiates Auto Blowdown.
2	RHR (LPCI) Pump Discharge Pressure Interlock	50 ± 10 psig	4 channels	Defers ADS actuation pending confirmation of Low Pressure core cooling system operation.
2	Core Spray Pump Discharge Pressure Interlock	185 ± 10 psig	4 channels	(LPCI or Core Spray Pump running interlock.)

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
1	RHR (LPCI) Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	NA	3 Inst. Channels	Monitors availability of power to logic systems.
1	HPCI Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.

LIMITING CONDITIONS FOR OPERATION

3.3.B Control Rods (Cont'd)

b. Whenever the reactor is in the startup or run modes below 25% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

c. If Specifications 3.3.B.3a or b cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 25%(*) rated power, it shall be brought to a shutdown condition immediately.

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

SURVEILLANCE REQUIREMENTS

4.3.B Control Rods (Cont'd)

b. 1. Prior to the start of control rod withdrawal towards criticality and prior to attaining 25% of rated power during rod insertion at shutdown, the Rod Worth Minimizer (RWM) shall be demonstrated to be operable by the following checks:

a. The RWM computer on line diagnostic test shall be successfully performed.

b. Prior to the start of control rod withdrawal only, proper annunciation of the selection error of at least one out-of-sequence control rod in a fully inserted group shall be verified.

c. The rod block function of the RWM shall be verified by withdrawing the first rod during start-up only as an out-of-sequence control rod no more than to the block point.

2. Following any loading of the rod worth minimizer sequence program into the computer, the correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.

c. When required, the presence of the second licensed operator to verify the following of the correct rod program shall be verified and recorded.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System.

Objective:

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification:

A. Normal System Availability

1. During periods when fuel is in the reactor and prior to startup from a Cold Condition, the Standby Liquid Control System shall be operable, except as specified in 3.4.B below. This system need not be operable when the reactor is in the Cold Condition and all control rods are fully inserted and Specification 3.3.A is met.

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal System Availability

The operability of the Standby Liquid Control System the performance of the following tests:

1. At least once per month each pump loop shall be functionally tested by recirculating demineralized water to the test tank.

2. At least once during each operating cycle:

(a) Check that the setting of the system relief valves is $1400 < P < 1680$ psig.

(b) Manually initiate the system, except explosive valves. Pump boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. Minimum pump flow rate of 39 gpm against a system head of 1225 psig shall be verified. After pumping boron solution the system will be flushed with demineralized water.

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM(Cont'd)

B. Operation with Inoperable Components

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM (Cont'd)

(c) Manually initiate one of the Standby Liquid Control Control System Pumps and pump demineralized water into the reactor vessel from the test tank.

This test checks explosion of the charge associated with the tested loop, proper operation of the explosive valves, and pump operability. The replacement charges to be installed will be selected from the same manufactured batch as the tested charge.

(d) Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

B. Surveillance with Inoperable Components

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily therafter until the inoperable component is repaired.

LIMITING CONDITIONS FOR OPERATION

3.5.A Core Spray and LPCI Subsystem (cont'd)

Both CSS shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Shutdown condition except as specified in 3.5.A.2 and 3.5.F.3 below:

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystems are operable.

SURVEILLANCE REQUIREMENTS

4.5.A Core Spray and LPCI Subsystem (cont'd)

	Item	Frequency
(d)	Pump Flow Rate	Once/3months
	Both loops shall deliver at least 6250 gpm against a system head corresponding to a reactor vessel pressure of 105 psig.	
(e)	Core Spray Header ∆P Instrumentation	
	Check	Once/day
	Calibrate	Once/3 months
(f)	Operability check to ensure that pumps will start and motor operated injection valves will open.	In accordance with 4.5.A.2, 4.5.A.4 and 4.5.A.5

2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem and the LPCI subsystems shall be demonstrated to be operable in accordance with 4.5.A.1(F) and 4.5.A.3(e) within 24 hours, and at least once per 72 hours thereafter until the inoperable core spray subsystem is restored to operable status.

3. LPCI Subsystem Testing shall be as follows:

	Item	Frequency
(a)	Simulated Automatic Actuation Test	Once/operating Cycle
(b)	Pump operability	Once/1 month

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.A Core Spray and LPCI Subsystem (cont'd)	4.5.A Core Spray and LPCI Subsystem (cont'd)

3.5.A Core Spray and LPCI Subsystem (cont'd)

3. Two independent Low Pressure Coolant Injection (LPCI) subsystems will be operable with each subsystem comprised of:

   a. (Two 33-1/3%) capacity pumps,

   b. An operable flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel, and

   c. During power operation the LPCI system cross-tie valve closed and the associated valve motor operator circuit breaker locked in the off position.

Both LPCI subsystems shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from the Cold Shutdown Condition, except as specified in 3.5.A.4 and 3.5.A.5 below.

4. From and after the date that one of the four LPCI pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days provided that during such seven days the remaining active components of the LPCI subsystems, and all active components of both core spray subsystems are operable.

5. From and after the date that one LPCI subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless it is sooner made operable, provided that during such 7 days all active components of both core spray subsystems and the remaining LPCI subsystem are operable.

4.5.A Core Spray and LPCI Subsystem (cont'd)

	Item	Frequency
(c)	Motor Operated valve operability	Once/month
(d)	Pump Flow Rate	Once/3 months
	Each LPCI pump shall deliver 10,900 gpm against a system head corresponding to a vessel pressure of 20 psig based on individual pump tests.	
(e)	Operability check to ensure that pumps will start and motor operated injection valves will open	In accordance with 4.5.A.2, 4.5.A.4 and 4.5.A.5

4. When it is determined that one of the RHR (LPCI) pumps is inoperable at a time when it is required to be operable, the remaining LPCI pumps and associated flow paths and both core spray subsystems shall be demonstrated to be operable in accordance with 4.5.1(f) and 4.5.A.3(e) within 24 hours and at least once per 72 hours thereafter until the LPCI subsystem is restored to operable status.

5. When it is determined that one of the LPCI subsystem is inoperable both core spray subsystems and the remaining LPCI subsystem shall be demonstrated to be operable within 24 hours, and at least once per 72 hours thereafter until the LPCI subsystem is restored to operabl status.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5.A Core Spray and LPCI Subsystem (cont'd)

6. All recirculation pump discharge valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).

7. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 48 hours.

B. Containment Cooling Subsystem (HPSW)

1. Except as specified in 3.5.B.2, 3.5.B.3, 3.5.B.4, and 3.5.F.3 below, all containment cooling subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, and prior to reactor startup from a Cold Shutdown Condition.

2. From and after the date that any two HPSW pumps are made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days, unless such pump is sooner made operable, provided that during such thirty days all active components of the containment cooling subsystem are operable.

4.5.A Core Spray and LPCI Subsystem (cont'd)

6. All recirculation pump discharge valves shall be tested for operabilty during any period of reactor cold shutdown exceedi[ng] 48 hours, if operability tests h[ave] not been performed during the preceeding 31 days.

B. Containment Cooling Subsystem (HPSW)

1. Containment Cooling Subsystem Testing shall be as follows:

	Item	Frequency
(a)	Pump Operability	Once/month
(b)	Motor operated valve operability	Once/month
(c)	Pump Capacity Test. Each HPSW pump shall deliver 4500 gpm at 280 psig.	After pump maintenance and every 3 months.
(d)	Air test on drywell and torus headers and nozzles.	Once/5 years

2. When it is determined that any two HPSW pumps are inoperable, the remaining components of the containment cooling subsystems shall be demonstrated to be operable immediately and weekly thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.7.D Primary Containment Isolation Valves

1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve shall be in the mode corresponding to the isolated condition.

4.7.D Primary Containment Isolation Valves

1. The primary containmen isolation valves surve lance shall be performed as llows:

   a. At least once per op rating cycle the operable i ola- tion valves that are ower operated and automat ally initiated shall be te ted for simulated automatic initiation and closure times.

   b. At least once per quarter:

      (1) All normally open power operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

      (2) With the reactor power less than 75% trip main steam isolatio valves individually a verify closure time.

   c. At least once per ek the main steam line po r- operated isolation alves shall be exercised y par- tial closure and su sequent reopening.

   d. At least once per o rating cycle the operabili of the reactor coolant ystem instrument line flo heck valves shall be veri ed.

2. Whenever an isolation valv listed in Table 3.7.1 is in er- able, the position of at le one other valve in each line having an inoperable valve shall be recorded daily.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3. ~~If Specificati~~ The reactor be initiated. the Cold Shutdown orderly shutdown shall **be initiated and the re- tor shall be in the ...d Shutdown condi... 24 hours within** unless Speci...ion 3.7.D.1 or 3.7.D can be met. condition within	

6.5.2 Operation and Safety Review Committee

Function

6.5.2.1 The Operation and Safety Review Committee shall function to provide independent review and audit of designated activities in the area of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

(the members of the OSR Committee will be competent in the area of quality assurance practice and cognizant of the Quality Assurance requirements of 10 CFR 50, Appendix B. Additionally, they will be cognizant of the corporate Quality Assurance Program and will have the corporate Quality Assurance Organization available to them.)

Composition

6.5.2.2 The Operation and Safety Review Committee shall be composed of the:

Manager-Electric Production Department (Chairman)
Superintendent-Maintenance Division
Superintendent-Services Division
Manager-Engineering & Research Department (Vice Chairman)
Chief Mechanical Engineer
Chief Electrical Engineer
Assistant Director-Research Division

Alternates

6.5.2.3 Alternate Members shall be appointed in writing by the OSR Committee Chairman. Each permanent member shall have a designated alternate to serve in his absence, and a current list of these alternates shall be maintained in Committee records. Each alternate member will serve on a continuing basis.

Consultants

6.5.2.4 Consultants shall be utilized as determined by the OSR Committee Chairman to provide expert advise to the OSR Committee.

Meeting Frequency

6.5.2.5 The OSR Committee shall meet at least once per six months.

Quorum

6.5.2.6 A quorum of the OSR Committee shall consist of the Chairman or Vice Chairman or their designated alternates and four members or their alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

Review

6.5.2.7 The OSR Committee shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

6.6 Reportable Occurrence Action

6.6.1 The following actions shall be taken in the event of a Reportable Occurrence:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each Reportable Occurrence Report submitted to the Commission shall be reviewed by the PORC and submitted to the OSR Committee and the Superintendent, Generation Division-Nuclear.

6.7 Safety Limit Violation

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36 (c) (1) (i) shall be complied with immediately.

b. The Safety Limit violation shall be reported to the Commission, the Superintendent, Generation Division-Nuclear or, in his absence, the Superintendent, Generation Division--Fossil-Hydro and to the OSR Committee immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the OSR Committee and the Superintendent, Generation Division-Nuclear within 14 days of the violation.

6.8 Procedures

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet the requirements of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 (November 1972) except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Station Superintendent or his designated alternate per Specification 6.1.1 prior to implementation and periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made, provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PORC and approved by the Station Superintendent within 14 days of implementation.

6.9 <u>Reporting Requirements</u>

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

6.9.1 <u>Routine Reports</u>

a. <u>Startup Report</u>. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hyraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be reported in this report.

6.9.1 Continued

c. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, and a narrative summary of the operating experience shall be submitted on a monthly basis to the Office of Management and Program Analysis (or its successor), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to be submitted no later than the 15th of the month following the calendar month covered by the report.

6.9.2 Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the orignal report date.

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate Region Office, or his designate no later than the first working day following the event, with a written followup report within ten working days. The written followup report shall shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

6.9.2 Continued

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in item 2.a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

6.9.3 Unique Reporting Requirements

Special reports shall be submitted to the Director of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Loss of shutdown margin, Specification 3.3.A and 4.3.A within 14 days of the event.

b. Reactor vessel inservice inspection, Specification 3.6.G and 4.6.G within 90 days of the completion of the reviews.

c. Secondary Containment leak rate testing, Specification 4.7.C within 90 days of the completion of the test.

d. Primary containment leak rate testing, Specification 4.7.A within 90 days of the completion of the test.

e. Release rate of Radioactive Effluents, Specification 3.8.B.7, 3.8.C.3.b, 3.8.C.5.

f. Sealed source leakage in excess of limits, Specification 3.13.3.

g. Effluent Releases

Effluent data should be summarized monthly, except in instances when more data is needed, and the items listed below reported semi-annually on the standard form "Report of Radioactive Effluents".

(1) Gaseous Releases

(a) Total radioactivity released (in curies) of noble and activation gases.

(b) maximum noble gas release rate during any one-hour period.

6.13 High Radiation Area

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c) (2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   2. A radiation monitoring device which continuously intergrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

   3. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over activities within the area and shall perform perdiodic radiation surveillance at the frequency specified by the plant Health Physicist or his designee on the Radiation Work Permit.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1 (a) above. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Superintendent or Shift Supervisor.

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. 47
License No. DPR-56

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

   A. The applications for amendment by Philadelphia Electric Company, et al, (the licensee) dated May 26 and September 5, 1978 comply 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 applications, 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. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 10, 1978

TTACHMENT TO LICENSE AMENDMENT NO. 47

CILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Revise Appendix A nical Specifications as follows:

Rem	Replace
iii	iii
6	67
6	68*
102	102a
115*	115*
116	116
12	125
126	126
12	127
17	177*
17	178
23	234k
23	234l*
23	234m*
24	249
25	250
25	253
25	254
25	254b
25	257
26	262
26	deleted
26	deleted

*No changes on thi ge

TABLE OF CONTENTS (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	Core Spray Pump Start Timer	6 ± 1 sec 10 ± 1 sec	4 timers 4 timers	In conjunction with loss of power initiates the starting of CSCS pumps.
1	LPCI Pump Start Timer	0 < t < 1 sec 5 ± 1 sec	2 timers 2 timers	This Specification shall be effective until replacement of the "0 < t < 1 sec" timers with auxiliary relays and addition of two "5 ± 1 sec" timers.
2	LPCI Pump Start Timer (Two pumps)	5 ± 1 sec	4 timers	This Specification shall be effective after replacement of the "0 < t < 1 sec" timers with auxiliary relays and addition of two "5 ± 1 sec" timers.
1	Auto Blowdown Timer	90 ≤ t ≤ 120	2 timers	In conjunction with Low Reactor Water Level, High Drywell Pressure and LPCI or Core Spray Pump running interlock, initiates Auto Blowdown.
2	RHR (LPCI) Pump Discharge Pressure Interlock	50 ± 10 psig	4 channels	Defers ADS actuation pending confirmation of Low Pressure core cooling system operation.
2	Core Spray Pump Discharge Pressure Interlock	185 ± 10 psig	4 channels	(LPCI or Core Spray Pump running interlock.)

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
1	RHR (LPCI) Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	NA	3 Inst. Channels	Monitors availability of power to logic systems.
1	HPCI Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	NA	2 Inst. Channels	Monitors availability of power to logic systems.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.B Control Rods (Cont'd)	4.3.B Control Rods (Cont'd)
b. Whenever the reactor is in the startup or run modes below 25% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.	b. 1. Prior to the start of control rod withdrawal towards criticality and prior to attaining 25% of rated power during rod insertion at shutdown, the Rod Worth Minimizer (RWM) shall be demonstrated to be operable by the following checks: a. The RWM computer on line diagnostic test shall be successfully performed. b. Prior to the start of control rod withdrawal only, proper annunciation of the selection error of at least one out-of-sequence control rod in a fully inserted group shall be verified. c. The rod block function of the RWM shall be verified by withdrawing the first rod during start-up only as an out-of-sequence control rod no more than to the block point. 2. Following any loading of the rod worth minimizer sequence program into the computer, the correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.
c. If Specifications 3.3.B.3a or b cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 21% rated power, it shall be brought to a shutdown condition immediately.	c. When required, the presence of the second licensed operator to verify the following of the correct rod program shall be verified and recorded.

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System.

Objective:

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification:

A. Normal System Availability

1. During periods when fuel is in the reactor and prior to startup from a Cold Condition, the Standby Liquid Control System shall be operable, except as specified in 3.4.B below. This system need not be operable when the reactor is in the Cold Condition and all control rods are fully inserted and Specification 3.3.A is met.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal System Availability

The operability of the Standby Liquid Control System the performance of the following tests:

1. At least once per month each pump loop shall be functionally tested by recirculating demineralized water to the test tank.

2. At least once during each operating cycle:

(a) Check that the setting of the system relief valves is $1400 < P < 1680$ psig.

(b) Manually initiate the system, except explosive valves. Pump boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. Minimum pump flow rate of 39 gpm against a system head of 1225 psig shall be verified. After pumping boron solution the system will be flushed with demineralized water.

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM (Cont'd)

B. Operation with Inoperable Components

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM (Cont'd)

(c) Manually initiate one of the Standby Liquid Control Control System Pumps and pump demineralized water into the reactor vessel from the test tank.

This test checks explosion of the charge associated with the tested loop, proper operation of the explosive valves, and pump operability. The replacement charges to be installed will be selected from the same manufactured batch as the tested charge.

(d) Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

B. Surveillance with Inoperable Components

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily therafter until the inoperable component is repaired.

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LIMITING CONDITIONS FOR OPERATION

3.5.A Core Spray and LPCI Subsystem (cont'd)

Both CSS shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Shutdown condition except as specified in 3.5.A.2 and 3.5.F.3 below:

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystems are operable.

SURVEILLANCE REQUIREMENTS

4.5.A Core Spray and LPCI Subsystem (cont'd)

	Item	Frequency
(d)	Pump Flow Rate	Once/3months
	Both loops shall deliver at least 6250 gpm against a system head corresponding to a reactor vessel pressure of 105 psig.	
(e)	Core Spray Header ΔP Instrumentation	
	Check	Once/day
	Calibrate	Once/3 months
(f)	Operability check to ensure that pumps will start and motor operated injection valves will open.	In accordance with 4.5.A.2, 4.5.A.4 and 4.5.A.5

2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem and the LPCI subsystems shall be demonstrated to be operable in accordance with 4.5.A.1(F) and 4.5.A.3(e) within 24 hours, and at least once per 72 hours thereafter until the inoperable core spray subsystem is restored to operable status.

3. LPCI Subsystem Testing shall be as follows:

	Item	Frequency
(a)	Simulated Automatic Actuation Test	Once/operating Cycle
(b)	Pump operability	Once/1 month

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5.A Core Spray and LPCI Subsystem (cont'd)

3. Two independent Low Pressure Coolant Injection (LPCI) subsystems will be operable with each subsystem comprised of:

   a. (Two 33-1/3%) capacity pumps,

   b. An operable flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel, and

   c. During power operation the LPCI system cross-tie valve closed and the associated valve motor operator circuit breaker locked in the off position.

Both LPCI subsystems shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from the Cold Shutdown Condition, except as specified in 3.5.A.4 and 3.5.A.5 below.

4. From and after the date that one of the four LPCI pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days provided that during such seven days the remaining active components of the LPCI subsystems, and all active components of both core spray subsystems are operable.

5. From and after the date that one LPCI subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless it is sooner made operable, provided that during such 7 days all active components of both core spray subsystems and the remaining LPCI subsystem are operable.

4.5.A Core Spray and LPCI Subsystem (cont'd)

	Item	Frequency
(c)	Motor Operated valve operability	Once/month
(d)	Pump Flow Rate	Once/3 months
	Each LPCI pump shall deliver 10,900 gpm against a system head corresponding to a vessel pressure of 20 psig based on individual pump tests.	
(e)	Operability check to ensure that pumps will start and motor operated injection valves will open	In accordance with 4.5.A.2, 4.5.A.4 and 4.5.A.5

4. When it is determined that one of the RHR (LPCI) pumps is inoperable at a time when it is required to be operable, the remaining LPCI pumps and associated flow paths and both core spray subsystems shall be demonstrated to be operable in accordance with 4.5.1(f) and 4.5.A.3(e) within 24 hours and at least once per 72 hours thereafter until the LPCI subsystem is restored to operable status.

5. When it is determined that one of the LPCI subsystem is inoperable both core spray subsystems and the remaining LPCI subsystem shall be demonstrated to be operable within 24 hours, and at least once per 72 hours thereafter until the LPCI subsystem is restored to operable status.

LIMITING CONDITIONS FOR OPERATION

3.5.A Core Spray and LPCI Subsystem (cont'd)

6. All recirculation pump discharge valves and bypass valve(s)[*] shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).

7. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 48 hours.

B. Containment Cooling Subsystem (HPSW)

1. Except as specified in 3.5.B.2, 3.5.B.3, 3.5.B.4, and 3.5.F.3 below, all containment cooling subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, and prior to reactor startup from a Cold Shutdown Condition.

2. From and after the date that any two HPSW pumps are made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days, unless such pump is sooner made operable, provided that during such thirty days all active components of the containment cooling subsystem are operable.

SURVEILLANCE REQUIREMENTS

4.5.A Core Spray and LPCI Subsystem (cont'd)

6. All recirculation pump discharge valves and bypass valve(s)[*] shall be tested for operability during any period of reactor cold shutdown exceeding 48 hours, if operability tests have not been performed during the preceeding 31 days.

B. Containment Cooling Subsystem (HPSW)

1. Containment Cooling Subsystem Testing shall be as follows:

	Item	Frequency
(a)	Pump Operability	Once/month
(b)	Motor operated valve operability	Once/month
(c)	Pump Capacity Test. Each HPSW pump shall deliver 4500 gpm at 280 psig.	After pump maintenance and every 3 months.
(d)	Air test on drywell and torus headers and nozzles.	Once/5 years

2. When it is determined that any two HPSW pumps are inoperable, the remaining components of the containment cooling subsystems shall be demonstrated to be operable immediately and weekly thereafter.

* Upon the removal of both recirculation pump discharge valve bypass valves, operability and surveillance of only the recirculation pump discharge valves is required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.D Primary Containment Isolation Valves	4.7.D Primary Containment Isolation Valves
1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.	1. The primary containment isolation valves surveillance shall be performed as follows: a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times. b. At least once per quarter: (1) All normally open power operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened. (2) With the reactor power less than 75% trip main steam isolation valves individually and verify closure time. c. At least once per week the main steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening. d. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.
2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve shall be in the mode corresponding to the isolated condition.	2. Whenever an isolation valve listed in Table 3.7.1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated. The reactor shall be in the Cold Shutdown condition within 24 hours unless Specification 3.7.D.1 or 3.7.D.2 can be met.	

TABLE 3.11.D.1 (Cont'd)

Safety Related Shock Suppressors (Snubbers)

SNUBBER NUMBER	LOCATION	ELEVATION	SNUBBER IN HIGH (1) RADIATION AREA DURING SHUTDOWN	SNUBBERS ESPECIALLY DIFFICULT TO REMOVE	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
10-GB-S-48	RHR	124	See 4.11.D.4.b			'B'RHR RM.
10-GB-S-49	RHR	124	"			"
10-GB-S-50	RHR	98	"			"
10-GB-S-51	RHR	98	"			'C'RHR RM.
10-GB-S-52	RHR	124	"			"
10-GB-S-53	RHR	124	"			"
10-GB-S-54	RHR	130	"	X		TORUS RM.
10-GB-S-55	RHR	130	"	X		"
10-GB-S-58	RHR	98	"			'B'RHR RM.
10-GB-S-62	RHR	102	"			'A'RHR RM.
10-GB-S-63	RHR	102	"			"
10-GB-S-64	RHR	93	"			"
10-GB-S-65	RHR	102	"			'D'RHR RM.
10-GB-S-66	RHR	102	"			"
10-GB-S-67	RHR	93	"			"
10-DCN-S-73	RHR	180	"	X	Drywell	
10-DCN-S-74	RHR	180	"	X	Drywell	

TABLE 3.11.D.1 (Cont'd)

Safety Related Shock Suppressors (Snubbers)

SNUBBER NUMBER	LOCATION	ELEVATION	SNUBBER IN HIGH(1) RADIATION AREA DURING SHUTDOWN	SNUBBERS ESPECIALLY DIFFICULT TO REMOVE	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
12-DCN-S-2	RWCU	173.5	See 4.11.D.4.b	X		RWCU ISOLATION VALVE RM. 165
12-DCN-S-5	RWCU	165	"	X	Drywell	
12-DCN-S-7	RWCU	165	"	X	Drywell	
14-DCN-5-23	CORE SPRAY	168	"	X	Drywell	
14-DCN-S-24	CORE SPRAY	168	"	X	Drywell	
14-DCN-S-26	CORE SPRAY	168	"	X	Drywell	
14-DCN-S-27	CORE SPRAY	168	"	X	Drywell	
13-HB-S-23	RCIC	103	"	X		RCIC ROOM
23-DBN-S-1	HPCI	121	"	X		TORUS ROOM
23-HB-S-1A	HPCI	103	"	X		HPCI ROOM
23-DDN-S-2A	HPCI	103	"			"
23-HB-S-3A	HPCI	100	"	X		"

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Safety Related Shock Suppressors (Snubbers)

SNUBBER NUMBER	LOCATION	ELEVATION	SNUBBER IN HIGH(1) RADIATION AREA DURING SHUTDOWN	SNUBBERS ESPECIALLY DIFFICULT TO REMOVE	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
23-DBN-S-6-1	HPCI	121	See 4.11.D.4.b	X		TORUS ROOM
23-DBN-S-6-2	HPCI	121	"	X		"
23-DBN-S-22	HPCI	155	"	X	Drywell	
23-DBN-S-23	HPCI	155	"	X	Drywell	
23-DDN-S-29	HPCI	117	"			HPCI ROOM
23-DDN-S-33	HPCI	93	"			"

Notes for Table 3.11.D.1

(1) Modifications to this Table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

Amendment No. 32

6.5.2 Operation and Safety Review Committee

Function

6.5.2.1 The Operation and Safety Review Committee shall function to provide independent review and audit of designated activities in the area of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

(the members of the OSR Committee will be competent in the area of quality assurance practice and cognizant of the Quality Assurance requirements of 10 CFR 50, Appendix B. Additionally, they will be cognizant of the corporate Quality Assurance Program and will have the corporate Quality Assurance Organization available to them.)

Composition

6.5.2.2 The Operation and Safety Review Committee shall be composed of the:

Manager-Electric Production Department (Chairman)
Superintendent-Maintenance Division
Superintendent-Services Division
Manager-Engineering & Research Department (Vice Chairman)
Chief Mechanical Engineer
Chief Electrical Engineer
Assistant Director-Research Division

Alternates

6.5.2.3 Alternate Members shall be appointed in writing by the OSR Committee Chairman. Each permanent member shall have a designated alternate to serve in his absence, and a current list of these alternates shall be maintained in Committee records. Each alternate member will serve on a continuing basis.

Consultants

6.5.2.4 Consultants shall be utilized as determined by the OSR Committee Chairman to provide expert advise to the OSR Committee.

Meeting Frequency

6.5.2.5 The OSR Committee shall meet at least once per six months.

Quorum

6.5.2.6 A quorum of the OSR Committee shall consist of the Chairman or Vice Chairman or their designated alternates and four members or their alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

Review

6.5.2.7 The OSR Committee shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

6.6 Reportable Occurrence Action

6.6.1 The following actions shall be taken in the event of a Reportable Occurrence:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each Reportable Occurrence Report submitted to the Commission shall be reviewed by the PORC and submitted to the OSR Committee and the Superintendent, Generation Division-Nuclear.

6.7 Safety Limit Violation

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36 (c) (1) (i) shall be complied with immediately.

b. The Safety Limit violation shall be reported to the Commission, the Superintendent, Generation Division-Nuclear or, in his absence, the Superintendent, Generation Division--Fossil-Hydro and to the OSR Committee immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the OSR Committee and the Superintendent, Generation Division-Nuclear within 14 days of the violation.

6.8 Procedures

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet the requirements of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 (November 1972) except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Station Superintendent or his designated alternate per Specification 6.1.1 prior to implementation and periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made, provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PORC and approved by the Station Superintendent within 14 days of implementation.

6.9 <u>Reporting Requirements</u>

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

6.9.1 <u>Routine Reports</u>

a. <u>Startup Report</u>. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hyraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be reported in this report.

6.9.1 Continued

c. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, and a narrative summary of the operating experience shall be submitted on a monthly basis to the Office of Management and Program Analysis (or its successor), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to be submitted no later than the 15th of the month following the calendar month covered by the report.

6.9.2 Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the orignal report date.

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate Region Office, or his designate no later than the first working day following the event, with a written followup report within ten working days. The written followup report shall shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

6.9.2 Continued

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in item 2.a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

6.9.3 Unique Reporting Requirements

Special reports shall be submitted to the Director of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Loss of shutdown margin, Specification 3.3.A and 4.3.A within 14 days of the event.

b. Reactor vessel inservice inspection, Specification 3.6.G and 4.6.G within 90 days of the completion of the reviews.

c. Secondary Containment leak rate testing, Specification 4.7.C within 90 days of the completion of the test.

d. Primary containment leak rate testing, Specification 4.7.A within 90 days of the completion of the test.

e. Release rate of Radioactive Effluents, Specification 3.8.B.7, 3.8.C.3.b, 3.8.C.5.

f. Sealed source leakage in excess of limits, Specification 3.13.3.

g. Effluent Releases

Effluent data should be summarized monthly, except in instances when more data is needed, and the items listed below reported semi-annually on the standard form "Report of Radioactive Effluents".

(1) Gaseous Releases

(a) Total radioactivity released (in curies) of noble and activation gases.

(b) maximum noble gas release rate during any one-hour period.

6.13 High Radiation Area

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c) (2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device which continuously intergrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

3. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over activities within the area and shall perform perdiodic radiation surveillance at the frequency specified by the plant Health Physicist or his designee on the Radiation Work Permit.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1 (a) above. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Superintendent or Shift Supervisor.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 47 AND 47 TO FACILITY LICENSE NOS. DPR-44 AND DPR-56

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

PEACH BOTTOM ATOMIC POWER STATION UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

Introduction

By letters dated May 26, and September 5, 1978, the Philadelphia Electric Company (the licensee) proposed changes to the Technical Specifications appended to Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Units Nos. 2 and 3. The changes relate to: (1) the instrumentation that initiates or controls the core and containment cooling systems, (2) Administrative Controls, (3) addition of safety related snubbers, (4) certain revisions that would conform to the staff's Standard Technical Specifications, and (5) various editorial changes to clarify the meaning or correct errors in the existing specifications.

Evaluation

1. Timing Relays

The licensee proposed changes to two types of timing relays in conjunction with the recommendations of the General Electric Company (GE) as described in Service Information Letter (SIL) No. 230, and SIL No. 230, Supplement No. 1, dated June 6, 1977 and December 30, 1977, respectively.

One of the proposed changes would revise the setting of the Automatic Depressurization System (ADS) timing relays from the present Technical Specification setting of 120 ± 5 seconds to ≤ 120 seconds. This change moves the upper setpoint limit to a lower, more conservative value. As indicated in Table 7.4.2 of the Peach Bottom Units Nos. 2 and 3 FSAR, only the upper limits of the ADS timer setting is used in the system analysis; therefore, the licensee requested that the lower limit be deleted.

We have reviewed the licensee's submittal and determined that the proposed revision to the upper setpoint limit is acceptable on the basis that it is being revised to a lower, more conservative value. However, during our review, we questioned the advisability of deleting the lower setpoint limit. As stated in Section 7.4.3.3.3 of the Peach Bottom FSAR, an ADS time delay "is chosen to be long enough so that the HPCIS has time to start,...". Thus, the basic purpose of the ADS time delay is not consistent with a zero setpoint. We discussed this with the licensee and recommended a lower setpoint of 90 seconds as recommended in GE SIL 230. The licensee agreed. Therefore, we find the licensee's proposal, as modified by the staff to be acceptable.

The second request related to timing relays involves the modification of the Low Pressure Coolant Injection (LPCI) system pump start circuits by replacing the presently installed "$0 < t < 1$ second" time delay relays, which are set at essentially zero seconds, with auxiliary relays. The "$0 < t < 1$ second" relays were provided by General Electric as part of their generic design.

Our review of the licensee's request and of the system design indicates that a time delay is not necessary for the Peach Bottom Units since the Peach Bottom design of the four LPCI pump start circuits consists of four separate circuits, one for each pump. Sequencing of the 4 pumps is not a part of the design. Therefore, replacing the timers with auxiliary relays will enhance system reliability. We find this requested change to be acceptable.

2. LPCI System Pump Start Circuits

Another of the licensee's requested changes involves the modification of the Low Pressure Coolant Injection (LPCI) System pump start circuits by the installation of additional time delay relays.

When the LPCI system recirculation loop selection logic was deleted in conjunction with Amendments Nos. 14 and 12 to License Nos. DPR-44 and DPR-56, respectively, the design of the LPCI logic was modified by adding a redundant signal to each pump start circuit. Further review of this design by the licensee indicated the desirability of adding four additional timers so that each pump start signal will have its own time delay relay. This improves system reliability by reducing the impact of certain postulated single failures.

Because of a licensee misunderstanding concerning the listing of the timers in Table 3.2.B, this modification was made on Unit No. 2 during the 1977 refueling outage without application for a Technical Specification change request. This event was reported to the NRC in LER 78-021/3L-0 (Unit 2). Accordingly, for Unit No. 2, the licensee requested that Table 3.2.B be modified by increasing the "Minimum Number of Operable Instrument Channels per Trip System" from "1" to "2" for both the "5 $\pm$ 1 second" and the "$0 < t < 1$ second" LPCI Pump Start Timers, and by increasing the "Number of Instrument Channels Provided by Design" from "2 timers" to "4 timers" for the same sets of timers.

We have reviewed the licensee's request and determined that these changes constitute a correction to conform the Technical Specifications to the installed design of the LPCI pump start circuits. Further we have determined that the installation of additional timers improves system reliability and is acceptable.

The above described modification has not been performed on Unit No. 3. The licensee proposed to perform this modification in conjunction with his request to replace $0 < t < 1$ second timers with auxiliary relays (as discussed in IIA above). Based on our foregoing discussion and evaluation, we find this change to be acceptable.

3. Editorial Correction to Table 3.2.B

The licensee requested a change to Table 3.2.B to correct an error in the Core Spray Pumps Starter Timing by changing the Minimum Number of Operable Instrument Channels per Trip System to "2" rather than "1" as previously listed. We have reviewed the licensee's submittal and determined that the revision will conform the Technical Specifications with the design of the Core Spray Pumps Starter Timing, as previously reviewed. This change is an editorial correction and is acceptable.

4. Administrative Controls

The licensee proposed changes to the Administrative Controls portion of the Technical Specifications. Some of the changes are editorial in nature whereas others would conform certain specifications to staff guidance as set forth in NUREG 0123 (Ref. 1). Each of these changes is discussed below:

a. The licensee proposed to change the titles of two members of the Operation and Safety Review (OSR) Committee to reflect changes in the management organization of the company.

We have reviewed the proposal and determined that since there are no changes to the qualifications of the membership, the revision is pro forma in nature and has no safety or environmental significance.

b. The licensee requested to delete the specific reference to the meeting frequency of the OSR Committee during the initial year of facility operation. Since this specific requirement is no longer applicable, we have determined the change to be editorial in nature and is acceptable.

c. The licensee proposed certain changes to clarify the specification involved: the definition of a quorum of the ORS Committee would add the Vice Chairman or his designated alternative as a member of the quorum; the November 1972 edition of Regulatory Guide 1.33 would be the specific reference for plant procedures; the approval mechanism and method for approval and periodic review of plant procedures would be revised to be consistent with other specifications and the Peach Bottom Operations Phase Quality Assurance Program which was previously approved by the staff (Ref. 2); the Commission's addressee for the Monthly Operating Report would be changed to reflect a title change within the NRC; the time period for submittal of the Primary and Secondary Containment leak rate test results would be specifically identified (consistent with 10 CFR 50, Appendix J, Section V.B.). We have reviewed each of the identified changes and determined that they are editorial in nature, more clearly identify staff approved procedures and requirements and are therefore acceptable.

d. The licensee proposed to modify the requirements for control of radiation areas by conforming this specification to the staff's published guidance (Ref. 1, Section 6.12). We have reviewed the licensee's submittal and determined that the proposal is consistent with the staff's previously approved alternative to the control device or alarm signal specified in paragraph 20.203(c)(2) of 10 CFR 20 and is acceptable.

5. Addition of Safety Related Shock Suppressors

The licensee proposed to add certain snubbers to the Table of Safety Related Shock Suppressors to reflect the recent addition of snubbers on Unit No. 3. This request is consistent with the current specifications as implemented by Amendment No. 32 to DPR-56, dated April 28, 1977, which states that snubbers may be added to safety related systems without prior approval provided the licensee proposes a revision to the Table. Therefore, we find the change to be an administrative action which implements a previously reviewed and approved amendment.

6. Rod Worth Minimizer

The licensee proposed to revise the surveillance requirements associated with the verification of the correctness of the control rod withdrawal sequence input to the Rod Worth Minimizer (RWM) computer. The revision would incorporate the staff's guidance (Ref. 1, paragraph 4.1.4.1.1.b) for verification following any loading of the program into the computer rather than the current specification which requires verification prior to each startup and shutdown.

The other surveillance requirements associated with RWM operability include diagnostic testing and verification of out-of-sequence rod withdrawal tests and associated annunciation. Therefore, we find that limiting the verification of the correctness of the input program to each loading operation is sufficient to provide adequate assurance of proper RWM operability.

7. Surveillance Testing of Valves and Pumps

The licensee proposed revision to certain of the specifications related to the surveillance requirements for the Standby Liquid Control System and the Core and Containment Cooling Systems. The changes would identify the specific type of valves involved and would increase the frequency of pump and valve operability tests for the Containment Cooling Subsystem from once/3 months to once/month to be consistent with testing requirements for other cooling systems.

The objective of surveillance testing of the cooling systems is to verify the operability of active components should they be required to respond to a facility abnormality. As such, the routine testing of manually operated valves is not a part of system response. This type of valve is routinely tested as part of the licensee's Inservice Inspection and Testing Program. Accordingly, we have determined that specifically identifying the type of valves (i.e., motor operated valves) is an editorial change to clarify the specification and is acceptable. The increased frequency of testing for the Containment Cooling Subsystem will provide additional assurance of operability and is also acceptable.

8. Recirculation Loop Discharge Valve Bypass Line

The licensee requested that the surveillance requirement for the recirculation loop discharge valve bypass line valve (RPDV BV) be deleted from the Unit No. 2 specifications and annotated for Unit No. 3 to provide the option to remove this bypass line at some future date. Prior to the issuance of Amendment No. 32 to DPR-44 dated February 24, 1977, the licensee removed the RPDV BP line. This modification was performed to eliminate the possibility of cracking in this line as was experienced in other Boiling Water Reactors. The surveillance requirements imposed by Amendment No. 32 resulted from the staff's review of the failure of RPDV or BV to close upon a Low Pressure Coolant Injection Signal. As stated in the Safety Evaluation supporting that Amendment, "We consider it necessary to require that surveillance be performed on the RPDV and BV (if installed)...". Accordingly, we find the proposed change to the Technical Specifications as a correction to the Unit No. 2 specification and an implementation of a previously reviewed and approved modification for Unit No. 3.

9. Staff Identified Change

During the course of the staff's review of the above described changes we identified a specification related to inoperable Primary Containment Isolation Valves (PCIV) that should be clarified. Specification 3.7.D.3 requires that if (1) a PCIV is inoperable and (2) the line having an inoperable valve cannot be placed in an isolated mode, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

As presently worded, the specification provides no capability to resume operation should the malfunction be cleared, i.e., the inoperable valve made operable or the line placed in an isolated mode before expiration of the stated 24 hour period. As stated in the staff's guidance for limiting conditions of operation (Ref. 1, Spec. 3.0.2), "In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required". Therefore, Specification 3.7.D.3 has been revised to provide this clarification. This change was discussed with the licensee and he agrees.

Environmental Considerations

We have determined that the amendments do not involve 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 Section 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusions

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of 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: October 10, 1978

REFERENCES:

1. NUREG - 0123, Rev. 1; Standard Technical Specifications for General Electric Boiling Water Reactors, April 1, 1978.

2. Letter, NRC (Lear) to PECo (Bauer) dated November 18, 1977.

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
AND
NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 47 and 47 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 revise the Technical Specifications which relate to: (1) the instrumentation that initiates or controls the core and containment cooling systems, (2) Administrative Controls, (3) addition of safety related snubbers, (4) certain revisions that would conform to the Staff's Standard Technical Specifications, and (5) various editorial changes to clarify the meaning or correct errors in the existing specifications.

The applications for the amendments comply 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 n 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 the amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 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 applications for amendments dated May 26 and September 5, 1978, (2) Amendments Nos. 47 and 47 to Licenses DPR-44 and DPR-56, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A 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 10 day of October 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

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