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November 19, 1985

Dockets Nos. 50-277
and 50-278

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

Dear Mr. Bauer:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENTS PERTAINING TO NUREG-0737 REQUIREMENTS

The Commission has issued the enclosed Amendments Nos. 113 and 117 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 Technical Specification (TS) changes pertaining to NUREG-0737 ("Clarification of TMI Action Plan Requirements") requirements in response to Generic Letter 83-36 ("NUREG-0737 Technical Specifications") sent to all BWR licensees on November 1, 1983. These changes were proposed in your amendment application dated October 9, 1984, as supplemented by a letter dated April 18, 1985. Your October 9, 1984, submittal referenced a February 11, 1982, amendment request which had been further amended on August 24, 1983. By amendments issued July 2, 1984, the Commission approved some of the changes requested in the February 11, 1982, submittal. Our current action involves certain outstanding items in your previous requests as well as your request of October 9, 1984, as supplemented.

These amendments cover the following TMI Action Plan items:

1. Post-Accident Sampling System (II.B.3)
2. Sampling and Analysis of Plant Effluents (II.F.1.2)
3. Noble Gas Effluent Monitors (II.F.1.1)
4. Containment High-Range Radiation Monitor (II.F.1.3)
5. Containment Pressure Monitor (II.F.1.4)
6. Containment Water Level Monitor (II.F.1.5)
7. Containment Hydrogen Monitor (II.F.1.6)
8. Control Room Habitability Requirements (III.D.3.4)
9. Switchover of Reactor Core Isolation Cooling (RCIC) Suction (II.K.3.22) (A TMI Action Plan item addressed previously in GL 83-02)
10. Instruments for Detection of Inadequate Core Cooling - Reactor Water Level Recorders (II.F.2)

The additional amendment request addressed in your October 9, 1984, application (SPDS - Reactor Pressure Recorder) is currently under staff review and will be handled as a separate plant specific action.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Biweekly Federal Register notice.

Sincerely,

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

Enclosures:
1. Amendment No. 113 to DPR-44
2. Amendment No. 117 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page

chgs made 10/30/85

w/noted changes

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
RIngram	GGears;cf	~~JStolz~~	L Finkelstein	GLainas
10/7/85	10/7/85	10/10/85	10/16/85	10/30/85

Mr. E. G. Bauer, Jr.
Philadelphia Electric Company

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Units 2 and 3

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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.113
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, et al. (the licensee) dated October 9, 1984, which supplemented a previous amendment application dated February 11, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 113, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective four months after issuance of this amendment except where otherwise noted.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 19, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 113

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

Remove	Insert
iii	iii
69	69
77	77
--	77a
78	78
--	78a
81a	81a
86	86
--	86a
93	93
93a	93a
130	130
172	172
173	173
193	193*
194	194
233	233
233a	233a
--	268
--	269

*Overleaf page included for document completeness.

TABLE OF CONTENTS (Cont'd)

Amendment No. 39, 47, 74, 75, 102, 104, 113

TABLE 3.2.B (CONTINUED)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Of Operable Instrument Channels Per Trip System	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1 (1)	Core Spray Sparger to Reactor Pressure Vessel d/p	1 (plus or minus 1.5) psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break.
2 (1)	Condensate Storage Tank Low Level	Greater than or equal to 5' above tank bottom	2 Inst. Channels	Provides interlock to HPCI pump suction valves.
2 (1)	Suppression Chamber High Level	Less than or equal to 5" above torus midpoint	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.
2 (6)	Condensate Storage Tank Low Level	Greater than or equal to 5' above tank bottom	2 Inst. Channels	Transfer RCIC pump suction to suppression chamber.

TABLE 3.2.F SURVEILLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	Action*
2	Reactor Water Level (narrow range)	Recorder 0-60" Indicator 0-60"	(1) (2) (3)
2	Reactor Water Level (wide range)	Recorder -165" to +50"	(10) (11)
2	Reactor Water Level (fuel zone)	Recorder -325" to 0"	(10) (11)
2	Reactor Pressure	Recorder 0-1500 psig Indicator 0-1200 psig	(1) (2) (3)
2	Drywell Pressure	Recorder 0-70 psig	(1) (2) (3)
2	Drywell Pressure (wide range)	Recorder 0-225 psig	(8) (9)
2	Drywell Pressure (subatmospheric range)	Recorder 5-25 psia	(8) (9)
2	Drywell Temperature	Recorder 0-400 degrees F Indicator 0-400 degrees F	(1) (2) (3)
2	Suppression Chamber Water Temperature	Recorder 30-230 degrees F Indicator 30-230 degrees F	(1) (2) (3) (6)
2	Suppression Chamber Water Level (narrow range)	Recorder 0-2 ft. Indicator 0-2 ft.	(1) (2) (3)

TABLE 3.2.F (Cont'd) - SURVEILLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	Action
2	Suppression Chamber Water Level (wide range)	Recorder 1-21 ft.	(10) (11)
1	Control Rod Position	28 Volt Indicating Lights	(1) (2) (3) (4)
1	Neutron Monitoring	SRM, IRM, LPRM 0-100%	(1) (2) (3) (4)
1	Safety-Relief Valve Position Indication	Acoustic or Thermocouple	(5)
2	Drywell High Range Radiation Monitors	Recorder 1-1E(+8) R/hr	(7)
1	Main Stack High Range Radiation Monitor	Recorder 1.4E(-2) to 1.4E(+4)uCi/cc	(7)
1	Reactor Building Roof Vent High Range Radiation Monitor	Recorder 1.4E(-2) to 1.4E(+4)uCi/cc	(7)
2	Drywell Hydrogen Concentration Analyzer and Monitor	Analyzer and Recorder 0-20% volume	(1) (2) (3)

* Notes for Table 3.2.F appear on pages 78 and 78a.

NOTES FOR TABLE 3.2.F

1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

3) If the requirements of notes (1) and (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

4) These surveillance instruments are considered to be redundant to each other.

5) If this parameter is not indicated in the control room, either restore at least one channel to operable status within thirty days or be in at least Hot Shutdown within the next 12 hours.

6) A suppression chamber water temperature instrument channel will be considered operable if there are at least ten (10) resistance temperature detector inputs operable and no two (2) adjacent resistance temperature detector inputs are inoperable.

7) With the number of operable channels less than the minimum number of instrument channels shown in Table 3.2.F, initiate the preplanned alternate method of monitoring the appropriate parameter within 72 hours and:

   a) either restore the inoperable channel(s) to operable status within 7 days of the event, or

   b) prepare and submit a Special Report to the Commission within 10 working days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

8) With the number of operable channels less than the minimum number of instrumentation channels shown in Table 3.2.F, continued operation is permissible during the succeeding thirty days, provided both Drywell Pressure instruments (0-70 psig) are operable; otherwise, restore the inoperable channel to operable status within 7 days or be in at least Hot Shutdown within the next 12 hours.

NOTES FOR TABLE 3.2.F (Cont'd)

9) If no channels are operable, continued operation is permissible during the succeeding 7 days, provided both Drywell Pressure instruments (0-70 psig) are operable; otherwise, restore the inoperable channel(s) to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

10) With the number of operable channels less than the minimum number of instrumentation channels shown in Table 3.2.F, continued operation is permissible during the succeeding 30 days, provided both narrow range instruments monitoring the same variable are operable; otherwise, restore the inoperable channel to operable status within 7 days or be in at least Hot Shutdown within the next 12 hours.

11) If no channels are operable, continued operation is permissible during the succeeding seven days, provided both narrow range instruments monitoring the same variable are operable; otherwise, restore the inoperable channel(s) to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

TABLE 4.2.B (CONTINUED)
MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
13) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None
14) HPCI Suction Source Levels	(1)	Once/3 months	None
15) 4KV Emergency Power System Voltage Relays (HGA,SV)	Once/operating cycle	Once/5 years	None
16) ADS Relief Valves Bellows Pressure Switches	Once/operating cycle	Once/operating cycle	None
17) LPCI/Cross Connect Valve Position	Once/refueling cycle	N/A	N/A
18) Condensate Storage Tank Level (RCIC) (7)	Once/3 months	Once/operating cycle	Once/day
19) 4KV Emergency Power Source Degraded Voltage Relays (IAV,CV-6,ITE)	Once/month	Once/operating cycle	None

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Water Level (narrow range)	Once/6 months	Once Each Shift
2) Reactor Water Level (wide range)	Once/operating cycle	Once/day
3) Reactor Water Level (fuel zone)	Once/operating cycle	Once/day
4) Reactor Pressure	Once/6 months	Once Each Shift
5) Drywell Pressure	Once/6 months	Once Each Shift
6) Wide Range Drywell Pressure	Once/operating cycle	Once/day
7) Subatmospheric Drywell Pressure	Once/operating cycle	Once/day
8) Drywell Temperature	Once/6 months	Once Each Shift
9) Suppression Chamber Water Temperature	Once/operating cycle	Once Each Day
10) Suppression Chamber Water Level	Once/6 months	Once Each Shift
11) Wide Range Suppression Chamber Water Level	Once/operating cycle	Once/day
12) Control Rod Position	NA	Once Each Shift
13) Neutron Monitoring (APRM)	Twice Per Week	Once Each Shift
14) Safety/Relief Valve Position Indicator (acoustics)	Once/operating cycle	Once/month
15) Safety/Relief Valve Position Indicator (thermocouple)	NA*	Once/month
16) Safety Valve Position Indicator (acoustics)	Once/operating cycle	Once/month
17) Safety Valve Position Indicator (thermocouple)	NA*	Once/month

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

	Instrument Channel	Calibration Frequency	Instrument Check
18)	Drywell High Range Radiation Monitors	Once/operating cycle** †	Once/month
19)	Main Stack High Range Radiation Monitor	Once/operating cycle †	Once/month
20)	Reactor Bldg. Roof Vent High Range Radiation Monitor	Once/operating cycle †	Once/month
21)	Drywell Hydrogen Concentration Analyzer and Monitor	Quarterly***	Once/month

* Perform instrument functional check once per operating cycle.

** Channel calibration shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/hr and a one point calibration check of the detector below 10R/hr with an installed or portable gamma source.

*** At least a two-point calibration using sample gas.

† This requirement becomes effective during the 1st refueling outage following Cycle 7 core reload.

3.2 BASES (Cont'd)

Four sets of two radiation monitors are provided which initiate the Reactor Building Isolation function and operation of the standby gas treatment system. Four instrument channels monitor the radiation from the refueling area ventilation exhaust ducts and four instrument channels monitor the building ventilation below the refueling floor. Each set of instrument channels is arranged in a 1 out of 2 twice trip logic.

Trip settings of less than 16 mr/hr for the monitors in the refueling area ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

Flow integrators are used to record the integrated flow of liquid from the drywell sumps. The integrated flow is indicative of reactor coolant leakage. A Drywell Atmosphere Radioactivity Monitor is provided to give supporting information to that supplied by the reactor coolant leakage monitoring system. (See Bases for 3.6.C and 4.6.C)

Some of the surveillance instrumentation listed in Table 3.2.F are required to meet the accident monitoring requirements of NUREG-0737, Clarification of TMI Action Plan Requirements. The instrumentation and the applicable NUREG-0737 requirements are:

1. Wide range drywell pressure (II.F.1.4)
2. Subatmospheric drywell pressure (II.F.1.4)
3. Wide range suppression chamber water level (II.F.1.5)
4. Main stack high range radiation monitor (II.F.1.1)
5. Reactor building roof vent high range radiation monitor (II.F.1.1)
6. Drywell hydrogen concentration analyzer and monitor (II.F.1.6)
7. Drywell high range radiation monitors (II.F.1.3)
8. Reactor Water Level - wide and fuel range (II.F.2)
9. Safety-Relief Valve position indication (II.D.3)

3.2 BASES (Cont'd.)

The recirculation pump trip has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event fall within the envelope of study events given in General Electric Company Topical Report, NEDO-10439, dated March, 1971.

In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200 degrees F. Restoration of the main steam line tunnel ventilation flow momentarily exposes the temperature sensors to high gas temperatures. The momentary temperature increase can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to increase the temperature trip setpoint to 250 degrees F for 30 minutes during restoration of ventilation system to avoid an unnecessary plant transient.

The Emergency Aux. Power Source Degraded Voltage trip function prevents damage to safety-related equipment in the event of a sustained period of low voltage. The voltage supply to each of the 4kV buses will be monitored by undervoltage relaying. With a degraded voltage condition on the off-site source, the undervoltage sensing relays operate to initiate a timing sequence.

The timing sequence provides constant and inverse time voltage characteristics. Degraded voltage protection includes: (1) An instantaneous relay (ITE) initiated at 90% voltage which initiates a 60-second time delay relay and a 6 second time delay relay. The 6-second time delay relay requires the presence of a safety injection signal to initiate transfer; (2) An inverse time voltage relay (CV-6) initiated at 87% voltage with a maximum 60 second delay and operates at 70% voltage in 30 seconds; and (3) An inverse time voltage relay (IAV) initiated at approximately 60% voltage and operates at 1.8 seconds at zero volts.

When the timing sequence is completed, the corresponding 4kV emergency circuit breakers are tripped and the emergency buses are transferred to the alternate source. The 60-second timing sequences were selected to prevent unnecessary transfers during motor starts and to allow the automatic tapchanger on the startup transformer to respond to the voltage condition. The 6-second timing sequence is necessary to prevent separation of the emergency buses from the off-site source during motor starting transients, yet still be contained within the time envelope in FSAR Table 8.5.1.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5.D Reactor Core Isolation Cooling (RCIC Sub-System)

1. The RCIC Sub-System shall be operable whenever there is irradiated fuel in the reactor vessel, the reactor pressure is greater than 105 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.D.2 below.

2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCIS is operable.

3. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 105 psig within 24 hours.

4.5.D Reactor Core Isolation Cooling (RCIC Sub-System)

1. RCIC Sub-System testing shall be performed as follows:

Item	Frequency
(a) Simulated Automatic Actuation Test*	Once/Operating Cycle
(b) Pump Operability	Once/Month
(c) Motor Operated Valve Operability	Once/Month
(d) Flow Rate at approximately 1000 psig Steam Pressure**	Once/3 Months
(e) Flow Rate at approximately 150 psig Steam Pressure**	Once/Operating Cycle
(f) Verify automatic transfer from CST to suppression pool on low CST water level	Once/Operating Cycle***

2. When it is determined that the RCIC sub-system is inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.

*Shall include automatic restart on low water level signal.

**The RCIC pump shall deliver at least 600 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

***Effective at 1st refueling outage after Cycle 7 reload.

3.7.A Primary Containment

6. Containment Atmosphere Dilution

a. Whenever either reactor is in power operation, the Post-LOCA Containment Atmosphere Dilution System must be operable and capable of supplying nitrogen to either Unit 2 or Unit 3 containment for atmosphere dilution if required by post-LOCA conditions. If this specification cannot be met, the system must be restored to an operable condition within 30 days or both reactors must be taken out of power operation.

b. Whenever either reactor is in power operation, the post-LOCA Containment Atmosphere Dilution System shall contain a minimum of 2000 gallons of liquid nitrogen. If this specification cannot be met, the minimum volume will be restored within 30 days or both reactors must be taken out of power operation.

c. Whenever either of the reactors is in power operation, there shall be at least one CAD system oxygen analyzer serving the drywell and one CAD system oxygen analyzer serving the suppression chamber on that reactor. If this specification cannot be met,

4.7.A Primary Containment

6. Containment Atmosphere Dilution

a. The post-LOCA containment atmosphere dilution system shall be functionally tested once per operating cycle.

b. The level in the liquid nitrogen storage tank shall be recorded weekly.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.7.A.6.c (Cont'd)

the unit shall be in Hot Shutdown within 12 hours.

d. A 30 psig limit is the maximum containment repressurization allowable using the CAD system. Venting via the SBGT system to this stack must be initiated at 30 psig following the initial peak pressure at 49.1 psig.

4.7.A.6 (Cont'd)

c. The CAD system oxygen analyzers shall be tested for operability using standard bottled oxygen once per month and shall be calibrated once per 6 months. The atmospheric analyzing system shall be functionally tested once per oprating cycle in conjunction with the specification 4.7.A.6.a. Should one of the two oxygen analyzers serving the drywell or suppression pool be found inoperable, the remaining analyzer serving the same compartment shall be tested for operability once per week until the defective analyzer is made operable.

3.7.A & 4.7.A BASES (Cont'd.)

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 0.5%/day at 56 psig. Calculations made by the AEC staff with leak rate and a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in TID 14844, show that the maximum total whole body passing cloud dose is about 1.0 REM and the maximum total thyroid dose is about 14 REM at 4500 meters from the stack over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population zone distance of 7300 meters are about 2.5 REM total whole body and 105 REM total thyroid. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

Drywell Interior

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

Post LOCA Atmosphere Dilution

In order to ensure that the containment atmosphere remains inerted, i.e. the oxygen-hydrogen mixture below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. During the first year of operation the normal inerting nitrogen makeup system will be available for this purpose. After that time the specifically designed CAD system will serve as the post-LOCA Containment Atmosphere Dilution System. By maintaining a minimum of 2000 gallons of liquid N_2 in the storage tank it is assured that a seven-day supply of N_2 for post-LOCA containment inerting is available. Since the inerting makeup system is continually functioning, no

3.7.A & 4.7.A BASES (Cont'd)

periodic testing of the system is required. Twice weekly operation of the containment oxygen analyzer that is associated with the containment inerting makeup system is sufficient to insure its readiness. Reliance on that oxygen analyzer for this purpose of post-LOCA oxygen measurement will terminate when the CAD system is operable.

The Post-LOCA Containment Atmosphere Dilution system design basis and description are presented in Question 14.6 of the FSAR. In summary, the limiting criteria, based on the assumptions of Safety Guide 7, are:

1. Maintain oxygen concentration in the containment during post-LOCA conditions to less than 5 % Volume.

2. Limit the buildup in the containment pressure due to nitrogen addition to less than 30 psig.

3. To limit the offsite dose due to containment venting (for pressure control) to less than 30 Rem to the thyroid.

By maintaining at least a 7-day supply of nitrogen on site, there will be sufficient time after the occurrence of a LOCA for obtaining additional nitrogen supply from local commercial sources which have been discussed in Question 14.6 of the FSAR. The system design contains sufficient redundancy to ensure its reliability. Thus, it is sufficient to test the operability of the whole system once per operating cycle. Redundant oxygen analyzers are provided for both the drywell and suppression chamber, i.e., there are four oxygen analyzers on each Unit. By permitting continued reactor operation at rated power with two of the four analyzers inoperable, redundancy of analyzing capability will be maintained while not imposing an unnecessary interruption in plant operation. If one of the two analyzers serving one of the compartments of the containment (drywell or suppression chamber) fails, the frequency of testing of the other analyzer of the same type serving the same compartment will be increased from monthly to weekly to assure its continued availability. Monthly testing of the analyzers using bottled oxygen will ensure the system's readiness because of the multiplicity of design. Since the analyzers are normally not in operation, there will be little deterioration due to use.

LIMITING CONDITIONS FOR OPERATION

3.11 Additional Safety Related Plant Capabilities

A. Main Control Room Emergency Ventilation System

1. Both control room emergency ventilation systems shall be operable at all times when secondary containment integrity is required except that one system may be out-of-service for 7 days.

2. If Specification 3.11.A.1 cannot be met, be in hot shutdown within 12 hours and cold shutdown within the following 24 hours.

3. With both control room emergency ventilation systems inoperable, suspend core alternations, handling of irradiated fuel in the secondary containment, and operations with a potential for draining the reactor vessel.

4. a. The results of the inplace cold DOP and halogenated hydrocarbon refrigerant tests at approximately 3,000 CFM on HEPA filters and charcoal adsorber filter trains shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal or that filter train shall not be considered operable.

SURVEILLANCE REQUIREMENTS

4.11 Additional Safety Related Plant Capabilities

A. Main Control Room Emergency Ventilation System

1. At least once per operating cycle, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8 inches of water at system design flow rate.

2. a. The tests and sample analysis of Specification 3.11.A.4 shall be performed initially and at least once per year for standby service; or after every 720 hours of system operation; or following significant painting, fire or chemical release in any ventilation zone communicating with the system while it is in operation.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter train or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon refrigerant testing shall be performed after each complete or partial replacement of the charcoal adsorber filters or after any structural maintenance on the system housing.

LIMITING CONDITIONS FOR OPERATION

b. The results of laboratory carbon sample analysis shall show 90% radioactive methyl iodide removal at a velocity within 20% of system design, 0.05 to 0.15 mg/m3 inlet methyl iodide concentration, $\geq$95% relative humidity and $\geq$125 degrees F, or that filter train shall not be considered operable.

c. Fans shall be shown to operate at approximately 3,000 CFM +/- 300 CFM (design flow for the filter train).

5. At least 1 of the 2 main control room intake air radiation monitors shall be operable with the inoperable channel failed safe whenever the control room emergency ventilation air supply fans and filter trains are required to be operable by 3.11.A.1 or filtration of the control room ventilation intake air must be initiated.

SURVEILLANCE REQUIREMENTS

d. A dry gas purge shall be provided to the filters to insure that the relative humidity in the filter systems does not exceed 70% during idle periods.

3. At least once per operating cycle automatic initiation of the control room air treatment system shall be demonstrated.

4. Operability of the main control room air intake radiation monitor shall be tested every 3 months.

6.19 Postaccident Sampling

Administrative controls shall be implemented to ensure the capability to obtain and analyze: (1) reactor coolant and containment atmosphere samples under accident conditions, and (2) radioactive iodines and particulates in plant gaseous effluents under accident conditions. The administrative controls shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis,
3. Provisions for maintenance of sampling and analytical equipment.

6.19 BASES

These administrative controls apply to the systems installed to ensure the capabilities required by NUREG-0737, Item II.B.3 (Post-Accident Sampling Capability) and Item II.F.1.2 (Iodine and Particulate Sampling).

The first capability is accomplished through the use of the Post-Accident Sampling System (PASS) located in the M-G set rooms and by the equipment available to handle, transport and analyze the samples. Analytical capability is provided at both the Unit 1 laboratory and an off-site laboratory, provided by contractual arrangements, for selected analyses. The off-site laboratory is relied upon to perform the chloride analysis required by NUREG-0737, Item II.B.3. The data obtained from the post-accident sampling system would be utilized to calculate the extent of fuel damage during accident conditons.

The second capability (II.F.1.2) is accomplished by the use of shielded sample collection devices, special handling tools, a shielded transport container, and high radiation measuring techniques. The collection devices (particulate filters and iodine cartridges) are located on the main stack and reactor building vent sampling systems.

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. 117
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, et al. (the licensee) dated October 9, 1984, which supplemented a previous amendment application dated February 11, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 117, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective four months after issuance of this amendment except where otherwise noted.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 19, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

Remove	Insert
iii	iii
69	69
77	77
--	77a
78	78
--	78a
81a	81a
86	86
--	86a
93	93
93a	93a
130	130
172	172
173	173
193	193*
194	194
233	233
233a	233a
--	268
--	269

*Overleaf page included for document completeness.

TABLE OF CONTENTS (Cont'd)

TABLE 3.2.F SU LLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	Action*
2	Reactor Water Level (narrow range)	Recorder 0-60" Indicator 0-60"	(1) (2) (3)
2	Reactor Water Level (wide range)	Recorder -165" to +50"	(10) (11)
2	Reactor Water Level (fuel zone)	Recorder -325" to 0"	(10) (11)
2	Reactor Pressure	Recorder 0-1500 psig Indicator 0-1200 psig	(1) (2) (3)
2	Drywell Pressure	Recorder 0-70 psig	(1) (2) (3)
2	Drywell Pressure (wide range)	Recorder 0-225 psig	(8) (9)
2	Drywell Pressure (subatmospheric range)	Recorder 5-25 psia	(8) (9)
2	Drywell Temperature	Recorder 0-400 degrees F Indicator 0-400 degrees F	(1) (2) (3)
2	Suppression Chamber Water Temperature	Recorder 30-230 degrees F Indicator 30-230 degrees F	(1) (2) (3) (6)
2	Suppression Chamber Water Level (narrow range)	Recorder 0-2 ft. Indicator 0-2 ft.	(1) (2) (3)

TABLE 3.2.B (CONTINUED)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Of Operable Instrument Channels Per Trip System	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1 (1)	Core Spray Sparger to Reactor Pressure Vessel d/p	1 (plus or minus 1.5) psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break.
2 (1)	Condensate Storage Tank Low Level	Greater than or equal to 5' above tank bottom	2 Inst. Channels	Provides interlock to HPCI pump suction valves.
2 (1)	Suppression Chamber High Level	Less than or equal to 5" above torus midpoint	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.
2 (6)	Condensate Storage Tank Low Level	Greater than or equal to 5' above tank bottom	2 Inst. Channels	Transfer RCIC pump suction to suppression chamber.

Amendment No. 74, 89, 112, 117

TABLE 3.2.F (Cont'd) - SURVEILLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	Action
2	Suppression Chamber Water Level (wide range)	Recorder 1-21 ft.	(10) (11)
1	Control Rod Position	28 Volt Indicating Lights	(1) (2) (3) (4)
1	Neutron Monitoring	SRM, IRM, LPRM 0-100%	(1) (2) (3) (4)
1	Safety-Relief Valve Position Indication	Acoustic or Thermocouple	(5)
2	Drywell High Range Radiation Monitors	Recorder 1-1E(+8) R/hr	(7)
1	Main Stack High Range Radiation Monitor	Recorder 1.4E(-2) to 1.4E(+4)uCi/cc	(7)
1	Reactor Building Roof Vent High Range Radiation Monitor	Recorder 1.4E(-2) to 1.4E(+4)uCi/cc	(7)
2	Drywell Hydrogen Concentration Analyzer and Monitor	Analyzer and Recorder 0-20% volume	(1) (2) (3)

* Notes for Table 3.2.F appear on pages 78 and 78a.

NOTES FOR TABLE 3.2.F

1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

3) If the requirements of notes (1) and (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

4) These surveillance instruments are considered to be redundant to each other.

5) If this parameter is not indicated in the control room, either restore at least one channel to operable status within thirty days or be in at least Hot Shutdown within the next 12 hours.

6) A suppression chamber water temperature instrument channel will be considered operable if there are at least ten (10) resistance temperature detector inputs operable and no two (2) adjacent resistance temperature detector inputs are inoperable.

7) With the number of operable channels less than the minimum number of instrument channels shown in Table 3.2.F, initiate the preplanned alternate method of monitoring the appropriate parameter within 72 hours and:

   a) either restore the inoperable channel(s) to operable status within 7 days of the event, or

   b) prepare and submit a Special Report to the Commission within 10 working days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

8) With the number of operable channels less than the minimum number of instrumentation channels shown in Table 3.2.F, continued operation is permissible during the succeeding thirty days, provided both Drywell Pressure instruments (0-70 psig) are operable; otherwise, restore the inoperable channel to operable status within 7 days or be in at least Hot Shutdown within the next 12 hours.

NOTES FOR TABLE 3.2.F (Cont'd)

9) If no channels are operable, continued operation is permissible during the succeeding 7 days, provided both Drywell Pressure instruments (0-70 psig) are operable; otherwise, restore the inoperable channel(s) to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

10) With the number of operable channels less than the minimum number of instrumentation channels shown in Table 3.2.F, continued operation is permissible during the succeeding 30 days, provided both narrow range instruments monitoring the same variable are operable; otherwise, restore the inoperable channel to operable status within 7 days or be in at least Hot Shutdown within the next 12 hours.

11) If no channels are operable, continued operation is permissible during the succeeding seven days, provided both narrow range instruments monitoring the same variable are operable; otherwise, restore the inoperable channel(s) to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

TABLE 4.2.B (CONTINUED)
MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
13) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None
14) HPCI Suction Source Levels	(1)	Once/3 months	None
15) 4KV Emergency Power System Voltage Relays (HGA,SV)	Once/operating cycle	Once/5 years	None
16) ADS Relief Valves Bellows Pressure Switches	Once/operating cycle	Once/operating cycle	None
17) LPCI/Cross Connect Valve Position	Once/refueling cycle	N/A	N/A
18) Condensate Storage Tank Level (RCIC) (7)	Once/3 months	Once/operating cycle	Once/day
19) 4KV Emergency Power Source Degraded Voltage Relays (IAV,CV-6,ITE)	Once/month	Once/operating cycle	None

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Water Level (narrow range)	Once/ 6 months	Once Each Shift
2) Reactor Water Level (wide range)	Once/operating cycle	Once/day
3) Reactor Water Level (fuel zone)	Once/operating cycle	Once/day
4) Reactor Pressure	Once/6 months	Once Each Shift
5) Drywell Pressure	Once/6 months	Once Each Shift
6) Wide Range Drywell Pressure	Once/operating cycle	Once/day
7) Subatmospheric Drywell Pressure	Once/operating cycle	Once/day
8) Drywell Temperature	Once/6 months	Once Each Shift
9) Suppression Chamber Water Temperature	Once/operating cycle	Once Each Day
10) Suppression Chamber Water Level	Once/6 months	Once Each Shift
11) Wide Range Suppression Chamber Water Level	Once/operating cycle	Once/day
12) Control Rod Position	NA	Once Each Shift
13) Neutron Monitoring (APRM)	Twice Per Week	Once Each Shift
14) Safety/Relief Valve Position Indicator (acoustics)	Once/operating cycle	Once/month
15) Safety/Relief Valve Position Indicator (thermocouple)	NA*	Once/month
16) Safety Valve Position Indicator (acoustics)	Once/operating cycle	Once/month
17) Safety Valve Position Indicator (thermocouple)	NA*	Once/month

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

	Instrument Channel	Calibration Frequency	Instrument Check
18)	Drywell High Range Radiation Monitors	Once/operating cycle** †	Once/month
19)	Main Stack High Range Radiation Monitor	Once/operating cycle †	Once/month
20)	Reactor Bldg. Roof Vent High Range Radiation Monitor	Once/operating cycle †	Once/month
21)	Drywell Hydrogen Concentration Analyzer and Monitor	Quarterly***	Once/month

* Perform instrument functional check once per operating cycle.

** Channel calibration shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/hr and a one point calibration check of the detector below 10R/hr with an installed or portable gamma source.

*** At least a two-point calibration using sample gas.

† This requirement becomes effective during the 1st refueling outage following the Cycle 7 core reload.

3.2 BASES (Cont'd)

Four sets of two radiation monitors are provided which initiate the Reactor Building Isolation function and operation of the standby gas treatment system. Four instrument channels monitor the radiation from the refueling area ventilation exhaust ducts and four instrument channels monitor the building ventilation below the refueling floor. Each set of instrument channels is arranged in a 1 out of 2 twice trip logic.

Trip settings of less than 16 mr/hr for the monitors in the refueling area ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

Flow integrators are used to record the integrated flow of liquid from the drywell sumps. The integrated flow is indicative of reactor coolant leakage. A Drywell Atmosphere Radioactivity Monitor is provided to give supporting information to that supplied by the reactor coolant leakage monitoring system. (See Bases for 3.6.C and 4.6.C)

Some of the surveillance instrumentation listed in Table 3.2.F are required to meet the accident monitoring requirements of NUREG-0737, Clarification of TMI Action Plan Requirements. The instrumentation and the applicable NUREG-0737 requirements are:

1. Wide range drywell pressure (II.F.1.4)
2. Subatmospheric drywell pressure (II.F.1.4)
3. Wide range suppression chamber water level (II.F.1.5)
4. Main stack high range radiation monitor (II.F.1.1)
5. Reactor building roof vent high range radiation monitor (II.F.1.1)
6. Drywell hydrogen concentration analyzer and monitor (II.F.1.6)
7. Drywell high range radiation monitors (II.F.1.3)
8. Reactor Water Level - wide and fuel range (II.F.2)

9. Safety-Relief Valve position indication (II.D.3)

3.2 BASES (Cont'd.)

The recirculation pump trip has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event fall within the envelope of study events given in General Electric Company Topical Report, NEDO-10439, dated March, 1971.

In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200 degrees F. Restoration of the main steam line tunnel ventilation flow momentarily exposes the temperature sensors to high gas temperatures. The momentary temperature increase can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to increase the temperature trip setpoint to 250 degrees F for 30 minutes during restoration of ventilation system to avoid an unnecessary plant transient.

The Emergency Aux. Power Source Degraded Voltage trip function prevents damage to safety-related equipment in the event of a sustained period of low voltage. The voltage supply to each of the 4kV buses will be monitored by undervoltage relaying. With a degraded voltage condition on the off-site source, the undervoltage sensing relays operate to initiate a timing sequence.

The timing sequence provides constant and inverse time voltage characteristics. Degraded voltage protection includes: (1) An instantaneous relay (ITE) initiated at 90% voltage which initiates a 60-second time delay relay and a 6 second time delay relay. The 6-second time delay relay requires the presence of a safety injection signal to initiate transfer; (2) An inverse time voltage relay (CV-6) initiated at 87% voltage with a maximum 60 second delay and operates at 70% voltage in 30 seconds; and (3) An inverse time voltage relay (IAV) initiated at approximately 60% voltage and operates at 1.8 seconds at zero volts.

When the timing sequence is completed, the corresponding 4kV emergency circuit breakers are tripped and the emergency buses are transferred to the alternate source. The 60-second timing sequences were selected to prevent unnecessary transfers during motor starts and to allow the automatic tapchanger on the startup transformer to respond to the voltage condition. The 6-second timing sequence is necessary to prevent separation of the emergency buses from the off-site source during motor starting transients, yet still be contained within the time envelope in FSAR Table 8.5.1.

LIMITING CONDITIONS FOR OPERATION

3.5.D Reactor Core Isolation Cooling (RCIC Sub-System)

1. The RCIC Sub-System shall be operable whenever there is irradiated fuel in the reactor vessel, the reactor pressure is greater than 105 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.D.2 below.

2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCIS is operable.

3. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 105 psig within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.D Reactor Core Isolation Cooling (RCIC Sub-System)

1. RCIC Sub-System testing shall be performed as follows:

Item	Frequency
(a) Simulated Automatic Actuation Test*	Once/Operating Cycle
(b) Pump Operability	Once/Month
(c) Motor Operated Valve Operability	Once/Month
(d) Flow Rate at approximately 1000 psig Steam Pressure**	Once/3 Months
(e) Flow Rate at approximately 150 psig Steam Pressure**	Once/Operating Cycle
(f) Verify automatic transfer from CST to suppression pool on low CST water level	Once/Operating*** Cycle

2. When it is determined that the RCIC sub-system is inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.

*Shall include automatic restart on low water level signal.

**The RCIC pump shall deliver at least 600 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

***Effective at 1st refueling outage after Cycle 7 reload.

3.7.A Primary Containment

6. Containment Atmosphere Dilution

a. Whenever either reactor is in power operation, the Post-LOCA Containment Atmosphere Dilution System must be operable and capable of supplying nitrogen to either Unit 2 or Unit 3 containment for atmosphere dilution if required by post-LOCA conditions. If this specification cannot be met, the system must be restored to an operable condition within 30 days or both reactors must be taken out of power operation.

b. Whenever either reactor is in power operation, the post-LOCA Containment Atmosphere Dilution System shall contain a minimum of 2000 gallons of liquid nitrogen. If this specification cannot be met, the minimum volume will be restored within 30 days or both reactors must be taken out of power operation.

c. Whenever either of the reactors is in power operation, there shall be at least one CAD system oxygen analyzer serving the drywell and one CAD system oxygen analyzer serving the suppression chamber on that reactor. If this specification cannot be met,

4.7.A Primary Containment

6. Containment Atmosphere Dilution

a. The post-LOCA containment atmosphere dilution system shall be functionally tested once per operating cycle.

b. The level in the liquid nitrogen storage tank shall be recorded weekly.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.7.A.6.c (Cont'd)

the unit shall be in Hot Shutdown within 12 hours.

d. A 30 psig limit is the maximum containment repressurization allowable using the CAD system. Venting via the SBGT system to this stack must be initiated at 30 psig following the initial peak pressure at 49.1 psig.

4.7.A.6 (Cont'd)

c. The CAD system oxygen analyzers shall be tested for operability using standard bottled oxygen once per month and shall be calibrated once per 6 months. The atmospheric analyzing system shall be functionally tested once per oprating cycle in conjunction with the specification 4.7.A.6.a. Should one of the two oxygen analyzers serving the drywell or suppression pool be found inoperable, the remaining analyzer serving the same compartment shall be tested for operability once per week until the defective analyzer is made operable.

3.7.A & 4.7.A BASES (Cont'd.)

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 0.5%/day at 56 psig. Calculations made by the AEC staff with leak rate and a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in TID 14844, show that the maximum total whole body passing cloud dose is about 1.0 REM and the maximum total thyroid dose is about 14 REM at 4500 meters from the stack over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population zone distance of 7300 meters are about 2.5 REM total whole body and 105 REM total thyroid. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

Drywell Interior

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

Post LOCA Atmosphere Dilution

In order to ensure that the containment atmosphere remains inerted, i.e. the oxygen-hydrogen mixture below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. During the first year of operation the normal inerting nitrogen makeup system will be available for this purpose. After that time the specifically designed CAD system will serve as the post-LOCA Containment Atmosphere Dilution System. By maintaining a minimum of 2000 gallons of liquid N_2 in the storage tank it is assured that a seven-day supply of N_2 for post-LOCA containment inerting is available. Since the inerting makeup system is continually functioning, no

3.7.A & 4.7.A BASES (Cont'd)

periodic testing of the system is required. Twice weekly operation of the containment oxygen analyzer that is associated with the containment inerting makeup system is sufficient to insure its readiness. Reliance on that oxygen analyzer for this purpose of post-LOCA oxygen measurement will terminate when the CAD system is operable.

The Post-LOCA Containment Atmosphere Dilution system design basis and description are presented in Question 14.6 of the FSAR. In summary, the limiting criteria, based on the assumptions of Safety Guide 7, are:

1. Maintain oxygen concentration in the containment during post-LOCA conditions to less than 5 % Volume.

2. Limit the buildup in the containment pressure due to nitrogen addition to less than 30 psig.

3. To limit the offsite dose due to containment venting (for pressure control) to less than 30 Rem to the thyroid.

By maintaining at least a 7-day supply of nitrogen on site, there will be sufficient time after the occurrence of a LOCA for obtaining additional nitrogen supply from local commercial sources which have been discussed in Question 14.6 of the FSAR. The system design contains sufficient redundancy to ensure its reliability. Thus, it is sufficient to test the operability of the whole system once per operating cycle. Redundant oxygen analyzers are provided for both the drywell and suppression chamber, i.e., there are four oxygen analyzers on each Unit. By permitting continued reactor operation at rated power with two of the four analyzers inoperable, redundancy of analyzing capability will be maintained while not imposing an unnecessary interruption in plant operation. If one of the two analyzers serving one of the compartments of the containment (drywell or suppression chamber) fails, the frequency of testing of the other analyzer of the same type serving the same compartment will be increased from monthly to weekly to assure its continued availability. Monthly testing of the analyzers using bottled oxygen will ensure the system's readiness because of the multiplicity of design. Since the analyzers are normally not in operation, there will be little deterioration due to use.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. The results of laboratory carbon sample analysis shall show 90% radioactive methyl iodide removal at a velocity within 20% of system design, 0.05 to 0.15 mg/m3 inlet methyl iodide concentration, $\geq$95% relative humidity and $\geq$125 degrees F, or that filter train shall not be considered operable.	d. A dry gas purge shall be provided to the filters to insure that the relative humidity in the filter systems does not exceed 70% during idle periods.
c. Fans shall be shown to operate at approximately 3,000 CFM +/- 300 CFM (design flow for the filter train).	3. At least once per operating cycle automatic initiation of the control room air treatment system shall be demonstrated.
5. At least 1 of the 2 main control room intake air radiation monitors shall be operable with the inoperable channel failed safe whenever the control room emergency ventilation air supply fans and filter trains are required to be operable by 3.11.A.1 or filtration of the control room ventilation intake air must be initiated.	4. Operability of the main control room air intake radiation monitor shall be tested every 3 months.

LIMITING CONDITIONS FOR OPERATION

3.11 Additional Safety Related Plant Capabilities

A. Main Control Room Emergency Ventilation System

1. Both control room emergency ventilation systems shall be operable at all times when secondary containment integrity is required except that one system may be out-of-service for 7 days.

2. If Specification 3.11.A.1 cannot be met, be in hot shutdown within 12 hours and cold shutdown within the following 24 hours.

3. With both control room emergency ventilation systems inoperable, suspend core alternations, handling of irradiated fuel in the secondary containment, and operations with a potential for draining the reactor vessel.

4. a. The results of the inplace cold DOP and halogenated hydrocarbon refrigerant tests at approximately 3,000 CFM on HEPA filters and charcoal adsorber filter trains shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal or that filter train shall not be considered operable.

SURVEILLANCE REQUIREMENTS

4.11 Additional Safety Related Plant Capabilities

A. Main Control Room Emergency Ventilation System

1. At least once per operating cycle, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8 inches of water at system design flow rate.

2. a. The tests and sample analysis of Specification 3.11.A.4 shall be performed initially and at least once per year for standby service; or after every 720 hours of system operation; or following significant painting, fire or chemical release in any ventilation zone communicating with the system while it is in operation.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter train or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon refrigerant testing shall be performed after each complete or partial replacement of the charcoal adsorber filters or after any structural maintenance on the system housing.

6.19 Postaccident Sampling

Administrative controls shall be implemented to ensure the capability to obtain and analyze: (1) reactor coolant and containment atmosphere samples under accident conditions, and (2) radioactive iodines and particulates in plant gaseous effluents under accident conditions. The administrative controls shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis,
3. Provisions for maintenance of sampling and analytical equipment.

6.19 BASES

These administrative controls apply to the systems installed to ensure the capabilities required by NUREG-0737, Item II.B.3 (Post-Accident Sampling Capability) and Item II.F.1.2 (Iodine and Particulate Sampling).

The first capability is accomplished through the use of the Post-Accident Sampling System (PASS) located in the M-G set rooms and by the equipment available to handle, transport and analyze the samples. Analytical capability is provided at both the Unit 1 laboratory and an off-site laboratory, provided by contractual arrangements, for selected analyses. The off-site laboratory is relied upon to perform the chloride analysis required by NUREG-0737, Item II.B.3. The data obtained from the post-accident sampling system would be utilized to calculate the extent of fuel damage during accident conditons.

The second capability (II.F.1.2) is accomplished by the use of shielded sample collection devices, special handling tools, a shielded transport container, and high radiation measuring techniques. The collection devices (particulate filters and iodine cartridges) are located on the main stack and reactor building vent sampling systems.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENTS NOS. 113 AND 117 TO FACILITY OPERATING LICENSES NOS. DPR-44 AND DPR-56

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

PEACH BOTTOM ATOMIC POWER STATION, UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

INTRODUCTION AND BACKGROUND

In November 1980, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements," which included all TMI Action Plan items approved by the Commission for implementation at nuclear power reactors. NUREG-0737 identifies those items for which Technical Specifications (TSs) are required. A number of items which require TSs were scheduled for implementation after December 31, 1981. The staff provided guidance on the scope of TSs for all of these items in Generic Letter 83-36. Generic Letter 83-36 was issued to all Boiling Water Reactor (BWR) licensees on November 1, 1983. In this Generic Letter, the staff requested licensees to:

1. Review their facility's TSs to determine if they were consistent with the guidance provided in the Generic Letter, and

2. Submit an application for a license amendment where deviations or absence of TSs were found.

By letter dated October 9, 1984, Philadelphia Electric Company (the licensee) responded to Generic Letter 83-36 by submitting a TS change request for Peach Bottom Units 2 and 3. This application amended a previous application dated February 11, 1982. This evaluation covers the following TMI Action Plan items:

1. Post-Accident Sampling System (II.B.3)
2. Sampling and Analysis of Plant Effluents (II.F.1.2)
3. Noble Gas Effluent Monitors (II.F.1.1)
4. Containment High-Range Radiation Monitor (II.F.1.3)
5. Containment Pressure Monitor (II.F.1.4)
6. Containment Water Level Monitor (II.F.1.5)
7. Containment Hydrogen Monitor (II.F.1.6)
8. Control Room Habitability Requirements (III.D.3.4)
9. Switchover of Reactor Core Isolation Cooling (RCIC) Suction (II.K.3.22)
10. Instruments for Detection of Inadequate Core Cooling - Reactor Water Level Recorders (II.F.2)

EVALUATION

1. Post-Accident Sampling (II.B.3)

The guidance provided by Generic Letter 83-36 requested that an administrative program should be established, implemented and maintained to ensure that the licensee has the capability to obtain and analyze reactor coolant and containment atmosphere samples under accident conditions. The Post-Accident Sampling System is not required to be operable at all times. Administrative procedures are to be established for returning inoperable instruments to operable status as soon as practicable.

The licensee has provided a proposed revision to the TSs which is consistent with the guidelines provided in Generic Letter 83-36. We conclude that the licensee has an acceptable TS for the Post-Accident Sampling System.

2. Noble Gas Effluent Monitors (II.F.1.1)

The licensee has supplemented the existing normal range monitors to provide noble gas monitoring in accordance with TMI Action Plan Item II.F.1.1. The proposed TSs for Noble Gas Effluent Monitors are consistent with the guidelines provided in Generic Letter 83-36. Therefore, we conclude that the TSs for Item II.F.1.1 are acceptable.

3. Sampling and Analysis of Plant Effluents (II.F.1.2)

The guidance provided by Generic Letter 83-36 requested that an administrative program should be established, implemented and maintained to ensure the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The licensee has proposed TSs that are consistent with our guidance. We conclude that the TSs for sampling and analysis of plant effluents are acceptable.

4. Drywell High-Range Radiation Monitor (II.F.1.3)

The licensee has installed two drywell radiation monitors in both Peach Bottom Units that are consistent with the guidance of TMI Action Plan Item II.F.1.3. Generic Letter 83-36 provided guidance for limiting conditions for operation and surveillance requirements for these monitors. The licensee proposed TSs that are consistent with the guidance provided in Generic Letter 83-36. Therefore, we conclude that the proposed TSs for Item II.F.1.3 are acceptable.

5. Drywell Pressure Monitor (II.F.1.4)

Both Peach Bottom Units have been provided with two wide range channels for monitoring drywell pressure following an accident. The licensee has proposed TSs that are consistent with the guidelines contained in Generic Letter 83-36. Therefore, we conclude that the proposed TSs for drywell pressure monitors are acceptable.

6. Suppression Chamber Water Level Monitor (II.F.1.5)

The suppression chamber wide range water level monitors at both Peach Bottom Units provide the capability required by TMI Action Plan Item II.F.1.5. The TSs for both units contain limiting conditions of operation and surveillance requirements that are consistent with the guidance contained in Generic Letter 83-36. Therefore, we conclude that the proposed TSs for suppression chamber wide range water level monitors are acceptable.

7. Drywell Hydrogen Monitor (II.F.1.6)

The licensee has installed drywell hydrogen monitors that provide the capability required by TMI Action Plan Item II.F.1.6. The licensee has proposed Technical Specifications for drywell hydrogen monitors which contain appropriate limiting conditions of operation and surveillance for these monitors. Additionally, the licensee has requested to remove existing specifications regarding these instruments from the TSs for containment atmosphere dilution systems as hydrogen monitors are included in the table for surveillance instrumentation. We have reviewed the proposed TSs and conclude that they are consistent with the guidance contained in Generic Letter 83-36. Therefore, we find the proposed TSs to be acceptable.

8. Control Room Habitability (III.D.3.4)

The guidance of NUREG-0737 requires assurance on the part of the licensee that control room operators will be adequately protected against the effects of an accidental release of toxic and/or radioactive gases from sources either onsite or offsite. Generic Letter 83-36 provided guidance on the toxic gas detection system, and a control room emergency air filtration system.

The TSs for control room emergency air treatment system are already included in existing TSs for the Peach Bottom Units 2 and 3. However, the licensee has proposed additional changes in the TSs to make them consistent with the guidance contained in Generic Letter 83-36. We have reviewed the proposed TSs and conclude that the proposed TSs are acceptable as they meet the intent of our guidance contained in Generic Letter 83-36. No toxic gas monitors are necessary at Peach Bottom Units 2 and 3.

9. Automatic Switchover of Reactor Core Isolation Cooling System (RCIC) Suction (II.K.3.22)

TMI Action Plan Item II.K.3.22 recommends modifications to the Reactor Core Isolation Cooling System (RCIC) such that RCIC system suction will automatically switchover from the condensate storage tank to the suppression pool when the condensate storage tank level is low.

In Generic Letter 83-02, the staff provided guidance on necessary changes in the TSs for implementation of the modifications. The proposed changes in TSs for RCIC are in response to Generic Letter 83-02. We have reviewed the proposed changes in the TSs and determined that the changes are consistent with the guidance provided in Generic Letter 83-02. We find these changes acceptable. TMI Action Plan Item II.K.3.13 was also addressed in Generic Letter 83-02, and the licensee proposed TSs which were approved by the NRC staff on July 2, 1984.

10. Instruments for Detection of Inadequate Core Cooling - Reactor Water Level Recorders (II.F.2)

The action statement for the narrow range water level instrument (indicator/recorder) on page 77 of the current Peach Bottom TSs requires a plant shutdown within 7 days if one channel is inoperable and shutdown within 48 hours if both channels are inoperable. The licensee proposes to increase the limiting condition for operation (LCO) to 30 days for one inoperable channel and 7 days for two inoperable channels. They justify the change by stating (1) the narrow range reactor level instruments are not safety related equipment, (2) safety-related level instruments are available, and (3) compensatory measures provided by the proposal strengthen the LCO for the wide and fuel range level instruments. We agree that the proposed change is acceptable.

The licensee proposes to strenghten the LCO action statement for the wide range and fuel zone range instrumentation as shown in the following table:

LCO SHUTDOWN COMPARISON

Reactor Water Level Instrument	One Inoperable Current	One Inoperable Proposal	Both Inoperable Current	Both Inoperable Proposal
Narrow Range (0-60 inches)	7 days	30 days	2 days	7 days
Wide Range (-165 to +50 inches)	Shutdown Not Required	30 days if both narrow range operable. 7 days if a narrow range inoperable.	30 days (2-11-82 application)	7 days if both narrow operable. 48 hours if a narrow range inoperable.

Fuel Zone (-325 to 0 inches)	Shutdown Not Required	30 days if both narrow range operable. 7 days if a narrow range inoperable.	30 days (2-11-82 application)	7 days if both narrow range operable. 48 hours if a narrow range inoperable.

The proposed changes should have a minor impact on availability. We find the proposed changes to be acceptable.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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: November 19, 1985

The following NRC personnel have contributed to this Safety Evaluation: C. Patel and W. Hodges