June 12, 1991

Docket Nos. 50-277
and 50-278

Mr. George J. Beck
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

DISTRIBUTION w/enclosures:

Docket File	ACRS(10)	WBateman, 17G-21
NRC PDR	GPA/PA	RBlough, R-I
Local PDR	OGC	LDoerflein, R-I
PDI-2 Rdg.	OC/LFMB	FRosa, 7E-4
SVarga	GHill (8)	
JCalvo	EJordan, 3701	
WButler	DHagan, 3206	
PMilano(2)	Wanda Jones, 7103	
RClark	RJones, 8E-23	
MO'Brien(2)	CMcCracken, 8D-1	

Dear Mr. Beck:

SUBJECT: CORE AND CONTAINMENT COOLING SYSTEMS SURVEILLANCE REQUIREMENTS FOR PEACH BOTTOM ATOMIC POWER STATION, UNIT NOS. 2 AND 3 (TAC NOS. 75961 AND 75962)

The Commission has issued the enclosed Amendment Nos. 160 and 162 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3. These amendments consist of changes to the Technical Specifications in response to your application dated January 30, 1990 as supplemented by letter dated April 9, 1991. The supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

These amendments revise the testing requirements for core and containment cooling systems when one system becomes inoperable, revise the operability requirements of the high pressure core cooling systems, and incorporate some administrative changes.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original signed by
Richard J. Clark

Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 160 to DPR-44
2. Amendment No. 162 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page
*Previously Concurred

OFC	:PDI-2/LA	:*PDI-2/PM	:PDI-2/PM	:*OTSB/BC	:*SRXB/BC	:*SELB	:*SPLB/BC	:*OGC	:PDI-2/D
NAME	:MO'Brien	:GSuh:tlc	:RClark	:JCalvo	:RJones	:FRosa	:CMcCracken	:STurk	:WButler
DATE	:6/11/91	:12/28/90	:06/13/91	:01/16/91	:01/24/91	:01/23/91	:02/15/91	:06/06/91	:6/13/91

OFFICIAL RECORD COPY
Document Name: TAC NOS. 75961/962

240039

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

June 12, 1991

Docket Nos. 50-277
and 50-278

Mr. George J. Beck
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Beck:

SUBJECT: CORE AND CONTAINMENT COOLING SYSTEMS SURVEILLANCE REQUIREMENTS FOR PEACH BOTTOM ATOMIC POWER STATION, UNIT NOS. 2 AND 3 (TAC NOS. 75961 AND 75962)

The Commission has issued the enclosed Amendment Nos. 160 and 162 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3. These amendments consist of changes to the Technical Specifications in response to your application dated January 30, 1990 as supplemented by letter dated April 9, 1991. The supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

These amendments revise the testing requirements for core and containment cooling systems when one system becomes inoperable, revise the operability requirements of the high pressure core cooling systems, and incorporate some administrative changes.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Richard J. Clark

Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 160 to DPR-44
2. Amendment No. 162 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. George J. Beck
Philadelphia Electric Company

Peach Bottom Atomic Power Station,
Units 2 and 3

cc:

J. W. Durham, Sr., Esquire
Sr. V.P. & General Counsel
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Philadelphia Electric Company
ATTN: Mr. D. B. Miller, Vice President
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

Philadelphia Electric Company
ATTN: Regulatory Engineer, A1-2S
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

Resident Inspector
U.S. Nuclear Regulatory Commission
Peach Bottom Atomic Power Station
P.O. Box 399
Delta, Pennsylvania 17314

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. Roland Fletcher
Department of Environment
201 West Preston Street
Baltimore, Maryland 21201

Single Point of Contact
P. O. Box 11880
Harrisburg, Pennsylvania 17108-1880

Mr. Thomas M. Gerusky, Director
Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Board of Supervisors
Peach Bottom Township
R. D. #1
Delta, Pennsylvania 17314

Public Service Commission of Maryland
Engineering Division
ATTN: Chief Engineer
231 E. Baltimore Street
Baltimore, MD 21202-3486

Mr. Richard McLean
Power Plant and Environmental
Review Division
Department of Natural Resources
B-3, Tawes State Office Building
Annapolis, Maryland 21401

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. 160
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 January 30, 1990, as supplemented by letter dated April 9, 1991, 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 or 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. 160, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 160

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 areas are indicated by marginal lines.

Remove	Insert
59	59
120	120
125	125
125a	---
126	126
128	128
128a	128a
128b	128b
129	129
130	130
131	131
134	134
135	135
136	136
138	138
139	139
141	141
205	205
209	209
210	210
214	214
216a-1	216a-1
216a-5	216a-5
240t	240t
254	254
256	256
257	257

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.2.D. Radiation Monitoring Systems-Isolation and Initiation Functions	4.2.D. Radiation Monitoring Systems-Isolation and Initiation Functions
1. Reactor Building Isolation and Standby Gas Treatment System The limiting conditions for operation are given in Table 3.2.D.	1. Reactor Building Isolation and Standby Gas Treatment System Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D. System logic shall be functionally tested as indicated in Table 4.2.D.
E. Drywell Leak Detection The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Section 3.6.C, "Coolant Leakage".	E. Drywell Leak Detection Instrumentation shall be calibrated and checked as indicated in Table 4.2.E.

3.4 BASES

STANDBY LIQUID CONTROL SYSTEM

The Standby Liquid Control System is also required to meet 10 CFR 50.62 (Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants). The Standby Liquid Control System must have the equivalent control capacity of an 86 gpm system of 13% weight natural sodium pentaborate in order to satisfy 10 CFR 50.62 requirements. This equivalency requirement is fulfilled by having a system which satisfies the equation given in 3.4.B.3. Each parameter (sodium pentaborate solution concentration, pump flow rate, and Boron-10 enrichment) is tested at an interval consistent with the potential for that parameter to vary and also to assure proper equipment performance. Boron-10 enrichment testing is only required when chemical addition occurs since change cannot occur by any process other than the addition of new chemicals to the Standby Liquid Control Solution Tank.

Normally, pre-mixed dry sodium pentaborate enriched in Boron-10 is added to demineralized water to form the solution. The pre-mixed sodium pentaborate is purchased with certification of its Boron-10 enrichment. The solution could be made by combining natural borax and Boron-10 enriched boric acid in stoichiometric quantities in demineralized water. Since both the borax and Boron-10 enriched boric acid have known Boron-10 enrichments, the resulting Boron-10 enriched sodium pentaborate also would have a known Boron-10 enrichment. This process is adequate for use in determining immediate compliance with 3.4.B.3 following chemical addition. The solution Boron-10 enrichment shall be subsequently verified by analysis to be acceptable.

The volume of solution stored is checked at a frequency to assure high reliability of the system. Solution level is indicated and alarmed in the control room.

C. Only one of the two Standby Liquid Control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired.

LIMITING CONDITIONS FOR OPERATION

3.5.A Core Spray & 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 subsystem are operable.

SURVEILLANCE REQUIREMENTS

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

Item		Frequency
(d)	Pump Flow Rate	Once/3 months
	Each pump in each loop shall deliver at least 3125 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)	DELETED	

2. DELETED

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

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.

SURVEILLANCE REQUIREMENTS

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)	DELETED	
4.	DELETED	
5.	DELETED	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Containment Cooling System (cont'd)	4.5.B Containment Cooling System (cont'd)
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 the remaining HPSW pumps are operable.	2. DELETED
3. From and after the date that any three HPSW pumps are made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding fifteen days unless such pumps are sooner made operable provided the remaining HPSW pump is operable.	3. DELETED
4a. The torus cooling mode of RHR shall be operable with two independent loops. Each loop consists of: (1) At least one operable RHR pump. (2) An operable flow path to pump water from the torus through an operable RHR heat exchanger and back to the torus via the flow test line. (3) An operable HPSW flow path through the operable heat exchanger associated with the operable RHR pump. b. With one torus cooling loop inoperable, restore the inoperable loop to operable status within seven days. c. With both torus cooling loops inoperable, restore at least one loop to operable status within eight hours.	4. DELETED

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Containment Cooling System (cont'd)	4.5.B Containment Cooling System (cont'd)
5a. The drywell spray mode of RHR shall be operable with two independent loops. Each loop consists of: (1) At least one operable RHR pump. (2) An operable flow path to pump water from the torus through an operable RHR heat exchanger to the drywell spray sparger. (3) An operable HPSW flow path through the operable heat exchanger associated with the operable RHR pump.	5. DELETED
b. With one drywell spray loop inoperable, restore the inoperable loop to operable status within seven days.	
c. With both drywell spray loops inoperable, restore at least one loop to operable status within eight hours.	
6a. The torus spray mode of RHR shall be operable with two independent loops. Each loop consists of: (1) At least one operable RHR pump. (2) An operable flow path to pump water from the torus through an operable RHR heat exchanger to the torus spray sparger. (3) An operable HPSW flow path through the operable heat exchanger associated with the operable RHR pump.	6. DELETED
b. With one torus spray loop inoperable, restore the inoperable loop to operable status within seven days.	
c. With both torus spray loops inoperable, restore at least one loop to operable status within eight hours.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Containment Cooling System (cont'd)	4.5.B Containment Cooling System (cont'd)
7. If the requirements of 3.5.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	
C. HPCI Subsystem	C. HPCI Subsystem
1. The HPCI Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, reactor steam pressure is greater than 105 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.C.2 and 3.5.C.3 below.	1. HPCI Subsystem testing shall be performed as follows:

Item	Frequency
(a) Simulated Automatic Actuation Test	Once/operating cycle

LIMITING CONDITIONS FOR OPERATION

3.5.C HPCI Subsystem (cont'd.)

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, providing that during such seven days all active components of the ADS subsystem, the RCIC system, the LPCI subsystem and both core spray subsystems are operable.

3. If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.C HPCI Subsystem (cont'd.)

Item	Frequency
(b) Pump Operability	Once/month
(c) Motor Operated Valve Operability	Once/month
(d) Flow Rate at 1000 psig Steam Pressure	Once/3 months
(e) Flow Rate at 150 psig Steam Pressure	Once/operating cycle

The HPCI pump shall deliver at least 5000 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

2. DELETED

LIMITING CONDITIONS FOR OPERATION

3.5.D Reactor Core Isolation Cooling (RCIC) Subsystem

1. The RCIC Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, the reactor steam 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 RCIC Subsystem 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 HPCI Subsystem 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) Subsystem

1. RCIC Subsystem 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. DELETED

*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.

LIMITING CONDITIONS FOR OPERATION

3.5.E Automatic Depressurization System (ADS)

1. The Automatic Depressurization Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor steam pressure is greater than 105 psig and prior to a startup from a Cold Condition, except as specified in 3.5.E.2 below.

2. From and after the date that one valve in the automatic depressurization subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such valve is sooner made operable, provided that during such seven days the HPCI subsystem is operable.

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

SURVEILLANCE REQUIREMENTS

4.5.E Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

   A simulated automatic actuation test shall be performed prior to startup after each refueling outage.

2. DELETED

3.5.A BASES

Core Spray and LPCI Subsystems

Core Spray Subsystem (CSS)

The CSS is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Two redundant loops each provide adequate core cooling capacity for all break sizes from 0.2 ft^2 up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the Automatic Depressurization System (ADS).

The CSS specifications are applicable whenever irradiated fuel is in the core because the CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and also provides a source for flooding of the core in case of accidental draining.

With one CSS inoperable, the verified operability (see 4.5 Bases) of the redundant full capacity CSS and the full capacity Low Pressure Coolant Injection system provides assurance of adequate core cooling and justifies the specified 7 days out-of-service period.

The surveillance requirements provide adequate assurance that the CSS will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

Low Pressure Coolant Injection System (LPCIS)

The LPCIS is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Two loops each with two pumps provide adequate core flooding for all break sizes from 0.2 ft^2 up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The LPCIS specifications are applicable whenever there is irradiated fuel in the reactor vessel because LPCIS is a primary source of water for flooding the core after the reactor vessel is depressurized.

With one LPCIS pump inoperable, or one LPCIS loop inoperable, adequate core flooding is assured by the verified operability (see 4.5 Bases) of the redundant LPCIS pumps or loop, and both CSS loops. The reduced redundancy justifies the specified 7 day out-of-service period.

The surveillance requirements provide adequate assurance that the LPCI will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

3.5.A BASES (Cont'd)

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate using the methods described in Reference (1). Using the results developed in this reference, the repair period is found to be 1/2 the test interval. This assumes that the core spray subsystems and LPCI constitute a 1 out of 3 system; however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 is 1 month.

Should one core spray subsystem become inoperable, the remaining core spray and the LPCI system are available should the need for core cooling arise. To assure that the remaining core spray and LPCI subsystems are available, they are verified to be operable (see 4.5 Bases).

Should the loss of one LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Two LPCI pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, which will be verified to be operable (see 4.5 Bases), a thirty day repair period is justified. If the LPCI subsystem is not available, at least 1 LPCI pump must be available to fulfill the containment cooling function. The 7 day repair period is set on this basis.

(1) Jacobs, I. M., "Guidelines for Determining Safe Test Intervals and Repairs Times for Engineered Safeguards", General Electric Co. A.P.E.D., April, 1969 (APED 5736)

3.5.B BASES

Containment Cooling System

The Peach Bottom Containment Cooling System consists of the High Pressure Service Water (HPSW) system and the drywell spray, torus spray and torus cooling modes of the Residual Heat Removal System (RHRS).

The torus cooling mode of RHR consists of two independent loops. A loop is defined as a flow path to pump water, with an RHR pump, from the torus through an RHR heat exchanger, then back to the torus via the flow test line. A flow path from an operable HPSW pump through that RHR heat exchanger completes the functional loop.

The drywell spray mode of RHR consists of two independent loops. A loop is defined as a flow path to pump water, with an RHR pump, from the torus through an RHR heat exchanger to the drywell spray sparger. A flow path from an operable HPSW pump through that RHR heat exchanger completes the functional loop.

The torus spray mode of RHR consists of two independent loops. A loop is defined as a flow path to pump water from the torus, with an RHR pump, through an RHR heat exchanger to the torus spray sparger. A flow path from an operable HPSW pump through that RHR heat exchanger completes the functional loop.

The design of these systems is predicated upon use of 1 RHR and 1 HPSW pump for heat removal after a design basis event. Thus, there are ample spares for margin above the design conditions. Loss of margin should be avoided and the equipment maintained in a state of operability so a 30-day out-of-service time is chosen for two HPSW pumps.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by verifying the operability (see 4.5 Bases) of the remaining cooling equipment.

3.5.C BASES (Cont'd)

The HPCI and RCIC as well as all other Core Standby Cooling Systems must be operable when starting up from a Cold Condition. It is realized that the HPCI and RCIC systems are not designed to operate until reactor pressure exceeds 150 psig and are automatically isolated before reactor pressure decreases below 100 psig. It is the intent of Specifications 3.5.C and 3.5.D to assure that when the reactor is being started up from a Cold Condition, the HPCI and RCIC Systems are not known to be inoperable.

D. RCIC System

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the main condenser is unavailable. The nuclear safety analysis, FSAR Appendix G, shows that RCIC also serves for decay heat removal when feed water is lost. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgements on the reliability of the HPCI system, an allowable repair time of 1 week is specified. Additional discussions on RCIC are included in the HPCI Bases above.

E. Automatic Depressurization System (ADS)

The limiting conditions for operating the ADS are derived from the Station Nuclear Operational Analysis (Appendix G) and a detailed functional analysis of the ADS (Section 6).

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS.

3.5.E BASES (Cont'd.)

With one ADS valve known to be incapable of automatic operation, four valves remain operable to perform their ADS function. However, since the ECCS Loss-of-Coolant Accident analysis for small line breaks assumed that all five ADS valves were operable, reactor operation with one ADS valve inoperable is only allowed to continue for seven (7) days provided that the HPCI system is verified to be operable and that the actuation logic for the (remaining) four ADS valves is verified to be operable (see 4.5 Bases).

F. Minimum Low Pressure Cooling and Diesel Generator Availability

The purpose of Specification F is to assure that adequate core cooling capability is available at all times. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Additionally, the specification provides minimum core flooding requirements during refueling operations. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI subsystem, HPCI, and RCIC are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. If a water hammer were to occur at the time at which the system were required, the system would still perform its design function. However, to minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition.

4.5 BASES

Core and Containment Cooling Systems Surveillance Frequencies

The performance of individual emergency core cooling systems (HPCI, LPCI, Core Spray and ADS) and the integrated performance of the emergency core cooling systems are described in analyses referenced in Section 6.5 of the Updated Final Safety Analysis Report. Periodic surveillance of pumps and valves is performed in accordance with ASME Code, Section XI, to the extent described in the Inservice Testing Plan, to verify that the systems will provide the flow rates required by the respective analyses. HPCI and RCIC flow tests are performed at two pressures so that the systems' capability to provide rated flow over their operating range is verified. HPSW flow tests verify that rated flow can be delivered to the RHR heat exchangers.

The testing interval for the core and containment cooling systems is based on industry practice, sound engineering judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with frequent tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining redundant cooling systems that the Limiting Conditions for Operation require to be operable during the allowable out-of-service time period. Verifying operability in this context means to administratively ensure that the remaining required systems or subsystems are not known to be inoperable (for example: confirming that equipment necessary for the systems or subsystems to perform their safety functions are not blocked out of service for maintenance). Performance of operability tests is not required.

4.5 I&J Surveillance Requirements Bases

Average and Local LHGR

The LHGR shall be checked daily to determine if fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow and only a few control rods are moved daily, a daily check of power distribution is adequate.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
to ≤ 6.0 mrem to the total body and to ≤ 20.0 mrem to any organ. When the calculated dose from the release of radioactive materials in liquid effluents exceeds any of the above limits, prepare and submit to the Commission within 21 working days, pursuant to Specification 6.9.2, a Special Report which identifies the causes for exceeding the limits and corrective actions that have been taken to reduce the releases of radioactive materials in liquid effluents and proposed corrective actions to be taken to assure that subsequent releases are within the limits. This Special Report shall also include (1) results of radiological analyses of the drinking water source and (2) the radiological impact on the potentially affected drinking water supplies with regard to 40 CFR 141, Safe Drinking Water Act. Reactor shutdown is not required.	
3. During release of radioactive wastes, the following conditions shall be met: a. The minimum dilution water required to satisfy 3.8.B.1 shall be met. b. The gross activity monitor and flow monitor on the waste effluent line shall be operable except as specified in	3a. The liquid radwaste effluents radiation monitor shall be calibrated every 12 months with a known radioactive source positioned in a reproducible geometry with respect to the sensor and every quarter by means of a source check. Additionally, an instrument functional test shall be performed every

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2. The air dose in areas at and beyond the SITE BOUNDARY (see Figure 3.8.1) due to noble gases in gaseous effluents released from the two reactors at the site shall be limited to the following: a. During any calendar quarter for gamma radiation: ≤ 10 mrad. During any calendar quarter for beta radiation: ≤ 20 mrad. b. During any calendar year for gamma radiation: ≤ 20 mrad. During any calendar year for beta radiation: ≤ 40 mrad. When the calculated air dose from radioactive noble gases in gaseous effluents exceeds any of the above limits, prepare and submit to the Commission within 21 working days, pursuant to Specification 6.9.2, a Special Report which identifies the causes for exceeding the limits and defines the corrective actions that have been taken to reduce the releases and proposed corrective actions to be taken to assure that subsequent releases will be within the above limits. Reactor shutdown is not required.	2. Cumulative dose contributions for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per month.
3. The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium and from all radionuclides in particulate form with	3. Cumulative dose contributions for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
half-lives greater than 8 days in gaseous effluents released from the two reactors at the site to areas at and beyond the SITE BOUNDARY (see Figure 3.8.1) shall be limited to the following: a. During any calendar quarter: ≤ 15 mrem. b. During any calendar year: ≤ 30 mrem. When the calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form, with half-lives greater than 8 days in gaseous effluents exceeds any of the above limits, prepare and submit to the Commission within 21 working days, pursuant to Specification 6.9.2, a Special Report. The report shall identify the causes for exceeding the limits and define the corrective actions that have been taken and proposed corrective actions to to assure that subsequent releases will be within the above limits. Reactor shutdown is not required.	shall be determined in accordance with the methodology and parameters in the ODCM at least once per month.
4. During release of gaseous wastes the following conditions shall be met to avoid exceeding the limits specified in 3.8.C.1: a. The main off-gas stack minimum dilution flow of 10,000 cfm shall be maintained. b. One reactor building exhaust vent monitor	4a. The reactor building exhaust vent and main stack noble gas radiation monitors shall be calibrated every 12 months with a known radioactive source positioned in a reproducible geometry with respect to the sensor, and every quarter by means of a functional test. The channel functional test

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

pursuant to Specification 6.9.2 a Special Report which includes the following information:

a. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems and the reason for its inoperability.
b. Action taken to restore the inoperable equipment to operable status.
c. Summary description of action taken to prevent a recurrence.

Reactor shutdown is not required.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

made, including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the limits have been exceeded. If such is the case, prepare and submit to the Commission, within 21 working days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and schedule for achieving conformance with the above limits. This Special Report shall include an analysis that estimates the radiation exposure to a MEMBER OF THE PUBLIC, including all effluent pathways and direct radiation, including the releases covered by this report, for the calendar year. It shall also describe levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations. If the estimated dose exceeds the above limits and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with 40 CFR 190. Submittal of the report is considered a timely request and a variance is granted until staff action on the request is complete.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Identify the new location in the next Radioactive Dose Assessment Report and include in the report revised figures and tables for the ODCM reflecting the new locations.	
3. Analyses shall be performed on radioactive materials supplied as part of the EPA Environmental Radioactivity Intercomparison Studies Program, or another Interlaboratory Comparison Program that has been approved by the Commission.	3a. A summary of the results obtained as part of the Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.2.
a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence in the Annual Radiological Environmental Operating Report.	
3.8.F Solid Radioactive Waste	4.8.F Solid Radioactive Waste
1. The solid radwaste system shall be used in accordance with a Process Control Program (PCP) to process wet radioactive wastes to meet shipping and burial ground requirements.	1. The PCP shall be used to ensure meeting the burial ground and shipping requirements prior to shipment of radioactive wastes from the site.
a. With the provisions of the Process Control Program not satisfied, suspend shipments of defectively packaged solid radioactive waste from the site. Reactor shutdown is not required.	

LIMITING CONDITIONS FOR OPERATION

3.15 Seismic Monitoring Instrumentation

Applicability

Applies to the operational status of the seismic monitoring instrumentation.

Specifications

A. The seismic monitoring instrumentation shown in Table 3.15 shall be operable.

B. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the NRC pursuant to Specification 6.9.2 within the next 10 working days outlining the cause of the malfunction and the plans for restoring the instrument(s) to operable status.

C. The provisions of Specification 3.0.c are not applicable.

SURVEILLANCE REQUIREMENTS

4.15 Seismic Monitoring Instrumentation

Applicability

Applies to the surveillance requirements of the seismic monitoring instrumentation.

Specifications

A. Each of the required seismic monitoring instruments shall be demonstrated operable by the performance of the Instrument Check, Instrument Functional Test, and Instrument Calibration operations at the frequencies shown in Table 4.15.

B. Each of the required seismic monitoring instruments actuated during a seismic event shall be restored to operable status within 24 hours and an Instrument Calibration performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the NRC pursuant to Specification 6.9.2 within the next 10 working days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

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 Plant Manager or his designated alternate per Specification 6.1.1 prior to implementation and reviewed 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 Plant Manager within 14 days of implementation.

6.8.4 Written procedures shall be established, implemented and maintained covering the activities of the radiological effluent technical specifications as referenced below:

a. Offsite Dose Calculation Manual

b. Quality Assurance Program for the environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.

6.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the NRC in accordance with 10 CFR 50.4, "Written Communications".

6.9.1 Routine Reports (cont'd)

c. Annual Safety/Relief Valve Report

Describe all challenges to the primary coolant system safety and relief valves. Challenges are defined as the automatic opening of the primary coolant safety or relief valves in response to high reactor pressure.

d. 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. Each report shall be submitted no later than the 15th of the month following the calendar month covered by the report.

e. Core Operating Limits Report

(1) Core operating limits shall be established and shall be documented in the CORE OPERATING LIMITS REPORT prior to each Operating Cycle, or prior to any remaining portion of an Operating Cycle, for the following:

a. The APLHGR for Specification 3.5.I,

b. The MCPR for Specification 3.5.K,

c. The K_f core flow adjustment factor for Specification 3.5.K,

d. The LHGR for Specification 3.5.J,

e. The upscale flow biased Rod Block Monitor setpoint and the upscale high flow clamped Rod Block monitor setpoint of Specification 3.2.C.

(2) The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents as amended and approved:

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version)

b. Philadelphia Electric Company Methodologies as described in:

(1) PECo-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code"

6.9.2 Unique Reporting Requirements

Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified herein 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. Report seismic monitoring instrumentation inoperable for more than 30 days (Specification 3.15.B) within the next 10 working days. Submit a seismic event analysis (Specification 4.15.B) within 10 working days of the event.

d. Primary containment leak rate testing approximately three months after the completion of the periodic integrated leak rate test (Type A) required by Specification 4.7.A.2.c.2. For each periodic test, leakage test results from Type A, B and C tests shall be reported. B and C tests are local leak rate tests required by Specification 4.7.A.2.f. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test.

e. Calculated dose from release of radioactive effluents, Specification 3.8.B.2, 3.8.B.4, 3.8.C.2, 3.8.C.3, 3.8.C.5, 3.8.D, and 3.8.E.1.b.

f. Sealed source leakage in excess of limits, Specification 3.13.2.

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. 162
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 January 30, 1990, as supplemented by letter dated April 9, 1991, 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 or 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. 162, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 162

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 areas are indicated by marginal lines.

Remove	Insert
59	59
120	120
125	125
125a	---
126	126
128	128
128a	128a
128b	128b
129	129
130	130
131	131
134	134
135	135
136	136
138	138
139	139
141	141
205	205
209	209
210	210
214	214
216a-1	216a-1
216a-5	216a-5
240t	240t
254	254
256	256
257	257

LIMITING CONDITIONS FOR OPERATION

3.2.D. Radiation Monitoring Systems-Isolation and Initiation Functions

1. Reactor Building Isolation and Standby Gas Treatment System

The limiting conditions for operation are given in Table 3.2.D.

E. Drywell Leak Detection

The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Section 3.6.C, "Coolant Leakage".

SURVEILLANCE REQUIREMENTS

4.2.D. Radiation Monitoring Systems-Isolation and Initiation Functions

1. Reactor Building Isolation and Standby Gas Treatment System

Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D.

System logic shall be functionally tested as indicated in Table 4.2.D.

E. Drywell Leak Detection

Instrumentation shall be calibrated and checked as indicated in Table 4.2.E.

3.4 BASES

STANDBY LIQUID CONTROL SYSTEM

The Standby Liquid Control System is also required to meet 10 CFR 50.62 (Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants). The Standby Liquid Control System must have the equivalent control capacity of an 86 gpm system of 13% weight natural sodium pentaborate in order to satisfy 10 CFR 50.62 requirements. This equivalency requirement is fulfilled by having a system which satisfies the equation given in 3.4.B.3. Each parameter (sodium pentaborate solution concentration, pump flow rate, and Boron-10 enrichment) is tested at an interval consistent with the potential for that parameter to vary and also to assure proper equipment performance. Boron-10 enrichment testing is only required when chemical addition occurs since change cannot occur by any process other than the addition of new chemicals to the Standby Liquid Control Solution Tank.

Normally, pre-mixed dry sodium pentaborate enriched in Boron-10 is added to demineralized water to form the solution. The pre-mixed sodium pentaborate is purchased with certification of its Boron-10 enrichment. The solution could be made by combining natural borax and Boron-10 enriched boric acid in stoichiometric quantities in demineralized water. Since both the borax and Boron-10 enriched boric acid have known Boron-10 enrichments, the resulting Boron-10 enriched sodium pentaborate also would have a known Boron-10 enrichment. This process is adequate for use in determining immediate compliance with 3.4.B.3 following chemical addition. The solution Boron-10 enrichment shall be subsequently verified by analysis to be acceptable.

The volume of solution stored is checked at a frequency to assure high reliability of the system. Solution level is indicated and alarmed in the control room.

C. Only one of the two Standby Liquid Control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired.

LIMITING CONDITIONS FOR OPERATION

3.5.A Core Spray & 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 subsystem are operable.

SURVEILLANCE REQUIREMENTS

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

Item		Frequency
(d)	Pump Flow Rate	Once/3 months

Each pump in each loop shall deliver at least 3125 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) DELETED

2. DELETED

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

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.

SURVEILLANCE REQUIREMENTS

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

Item		Frequency
(c)	Motor Operated valve operability	Once/month
(d)	Pump Flow Rate 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.	Once/3 months
(e)	DELETED	
4.	DELETED	
5.	DELETED	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Containment Cooling System (cont'd)	4.5.B Containment Cooling System (cont'd)
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 the remaining HPSW pumps are operable.	2. DELETED
3. From and after the date that any three HPSW pumps are made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding fifteen days unless such pumps are sooner made operable provided the remaining HPSW pump is operable.	3. DELETED
4a. The torus cooling mode of RHR shall be operable with two independent loops. Each loop consists of: (1) At least one operable RHR pump. (2) An operable flow path to pump water from the torus through an operable RHR heat exchanger and back to the torus via the flow test line. (3) An operable HPSW flow path through the operable heat exchanger associated with the operable RHR pump.	4. DELETED
b. With one torus cooling loop inoperable, restore the inoperable loop to operable status within seven days.	
c. With both torus cooling loops inoperable, restore at least one loop to operable status within eight hours.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.B Containment Cooling System (cont'd)	4.5.B Containment Cooling System (cont'd)
5a. The drywell spray mode of RHR shall be operable with two independent loops. Each loop consists of: (1) At least one operable RHR pump. (2) An operable flow path to pump water from the torus through an operable RHR heat exchanger to the drywell spray sparger. (3) An operable HPSW flow path through the operable heat exchanger associated with the operable RHR pump.	5. DELETED
b. With one drywell spray loop inoperable, restore the inoperable loop to operable status within seven days.	
c. With both drywell spray loops inoperable, restore at least one loop to operable status within eight hours.	
6a. The torus spray mode of RHR shall be operable with two independent loops. Each loop consists of: (1) At least one operable RHR pump. (2) An operable flow path to pump water from the torus through an operable RHR heat exchanger to the torus spray sparger. (3) An operable HPSW flow path through the operable heat exchanger associated with the operable RHR pump.	6. DELETED
b. With one torus spray loop inoperable, restore the inoperable loop to operable status within seven days.	
c. With both torus spray loops inoperable, restore at least one loop to operable status within eight hours.	

LIMITING CONDITIONS FOR OPERATION

3.5.B Containment Cooling System (cont'd)

7. If the requirements of 3.5.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

C. HPCI Subsystem

1. The HPCI Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, reactor steam pressure is greater than 105 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.C.2 and 3.5.C.3 below.

SURVEILLANCE REQUIREMENTS

4.5.B Containment Cooling System (cont'd)

C. HPCI Subsystem

1. HPCI Subsystem testing shall be performed as follows:

	Item	Frequency
(a)	Simulated Automatic Actuation Test	Once/operating cycle

LIMITING CONDITIONS FOR OPERATION

3.5.C HPCI Subsystem (cont'd.)

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, providing that during such seven days all active components of the ADS subsystem, the RCIC system, the LPCI subsystem and both core spray subsystems are operable.

3. If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.C HPCI Subsystem (cont'd.)

Item	Frequency
(b) Pump Operability	Once/month
(c) Motor Operated Valve Operability	Once/month
(d) Flow Rate at 1000 psig Steam Pressure	Once/3 months
(e) Flow Rate at 150 psig Steam Pressure	Once/operating cycle

The HPCI pump shall deliver at least 5000 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

2. DELETED

LIMITING CONDITIONS FOR OPERATION

3.5.D Reactor Core Isolation Cooling (RCIC) Subsystem

1. The RCIC Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, the reactor steam 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 RCIC Subsystem 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 HPCI Subsystem 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) Subsystem

1. RCIC Subsystem 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. DELETED

*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.

LIMITING CONDITIONS FOR OPERATION

3.5.E Automatic Depressurization System (ADS)

1. The Automatic Depressurization Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor steam pressure is greater than 105 psig and prior to a startup from a Cold Condition, except as specified in 3.5.E.2 below.

2. From and after the date that one valve in the automatic depressurization subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such valve is sooner made operable, provided that during such seven days the HPCI subsystem is operable.

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

SURVEILLANCE REQUIREMENTS

4.5.E Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

   A simulated automatic actuation test shall be performed prior to startup after each refueling outage.

2. DELETED

3.5.A BASES

Core Spray and LPCI Subsystems

Core Spray Subsystem (CSS)

The CSS is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Two redundant loops each provide adequate core cooling capacity for all break sizes from 0.2 ft^2 up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the Automatic Depressurization System (ADS).

The CSS specifications are applicable whenever irradiated fuel is in the core because the CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and also provides a source for flooding of the core in case of accidental draining.

With one CSS inoperable, the verified operability (see 4.5 Bases) of the redundant full capacity CSS and the full capacity Low Pressure Coolant Injection system provides assurance of adequate core cooling and justifies the specified 7 days out-of-service period.

The surveillance requirements provide adequate assurance that the CSS will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

Low Pressure Coolant Injection System (LPCIS)

The LPCIS is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Two loops each with two pumps provide adequate core flooding for all break sizes from 0.2 ft^2 up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The LPCIS specifications are applicable whenever there is irradiated fuel in the reactor vessel because LPCIS is a primary source of water for flooding the core after the reactor vessel is depressurized.

With one LPCIS pump inoperable, or one LPCIS loop inoperable, adequate core flooding is assured by the verified operability (see 4.5 Bases) of the redundant LPCIS pumps or loop, and both CSS loops. The reduced redundancy justifies the specified 7 day out-of-service period.

The surveillance requirements provide adequate assurance that the LPCI will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

3.5.A BASES (Cont'd)

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate using the methods described in Reference (1). Using the results developed in this reference, the repair period is found to be 1/2 the test interval. This assumes that the core spray subsystems and LPCI constitute a 1 out of 3 system; however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 is 1 month.

Should one core spray subsystem become inoperable, the remaining core spray and the LPCI system are available should the need for core cooling arise. To assure that the remaining core spray and LPCI subsystems are available, they are verified to be operable (see 4.5 Bases).

Should the loss of one LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Two LPCI pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, which will be verified to be operable (see 4.5 Bases), a thirty day repair period is justified. If the LPCI subsystem is not available, at least 1 LPCI pump must be available to fulfill the containment cooling function. The 7 day repair period is set on this basis.

(1) Jacobs, I. M., "Guidelines for Determining Safe Test Intervals and Repairs Times for Engineered Safeguards", General Electric Co. A.P.E.D., April, 1969 (APED 5736)

3.5.B BASES

Containment Cooling System

The Peach Bottom Containment Cooling System consists of the High Pressure Service Water (HPSW) system and the drywell spray, torus spray and torus cooling modes of the Residual Heat Removal System (RHRS).

The torus cooling mode of RHR consists of two independent loops. A loop is defined as a flow path to pump water, with an RHR pump, from the torus through an RHR heat exchanger, then back to the torus via the flow test line. A flow path from an operable HPSW pump through that RHR heat exchanger completes the functional loop.

The drywell spray mode of RHR consists of two independent loops. A loop is defined as a flow path to pump water, with an RHR pump, from the torus through an RHR heat exchanger to the drywell spray sparger. A flow path from an operable HPSW pump through that RHR heat exchanger completes the functional loop.

The torus spray mode of RHR consists of two independent loops. A loop is defined as a flow path to pump water from the torus, with an RHR pump, through an RHR heat exchanger to the torus spray sparger. A flow path from an operable HPSW pump through that RHR heat exchanger completes the functional loop.

The design of these systems is predicated upon use of 1 RHR and 1 HPSW pump for heat removal after a design basis event. Thus, there are ample spares for margin above the design conditions. Loss of margin should be avoided and the equipment maintained in a state of operability so a 30-day out-of-service time is chosen for two HPSW pumps.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by verifying the operability (see 4.5 Bases) of the remaining cooling equipment.

3.5 BASES (Cont'd).

The HPCI and RCIC as well as all other Core Standby Cooling Systems must be operable when starting up from a Cold Condition. It is realized that the HPCI and RCIC systems are not designed to operate until reactor pressure exceeds 150 psig and are automatically isolated before reactor pressure decreases below 100 psig. It is the intent of Specifications 3.5.C and 3.5.D to assure that when the reactor is being started up from a Cold Condition, the HPCI and RCIC Systems are not known to be inoperable.

D. RCIC System

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the main condenser is unavailable. The nuclear safety analysis, FSAR Appendix G, shows that RCIC also serves for decay heat removal when feed water is lost. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgements on the reliability of the HPCI system, an allowable repair time of 1 week is specified. Additional discussions on RCIC are included in the HPCI Bases above.

E. Automatic Depressurization System (ADS)

The limiting conditions for operating the ADS are derived from the Station Nuclear Operational Analysis (Appendix G) and a detailed functional analysis of the ADS (Section 6).

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI or Core Spray and is based on 4 valves. There are five valves provided.

3.5.E BASES (Cont'd.)

With one ADS valve known to be incapable of automatic operation, four valves remain operable to perform their ADS function. However, since the ECCS Loss-of-Coolant Accident analysis for small line breaks assumed that all five ADS valves were operable, reactor operation with one ADS valve inoperable is only allowed to continue for seven (7) days provided that the HPCI system is verified to be operable and that the actuation logic for the (remaining) four ADS valves is verified to be operable (see 4.5 Bases).

F. Minimum Low Pressure Cooling and Diesel Generator Availability

The purpose of Specification F is to assure that adequate core cooling capability is available at all times. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Additionally, the specification provides minimum core flooding requirements during refueling operations. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI subsystem, HPCI, and RCIC are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. If a water hammer were to occur at the time at which the system were required, the system would still perform its design function. However, to minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition.

4.5 BASES

Core and Containment Cooling Systems Surveillance Frequencies

The performance of individual emergency core cooling systems (HPCI, LPCI, Core Spray and ADS) and the integrated performance of the emergency core cooling systems are described in analyses referenced in Section 6.5 of the Updated Final Safety Analysis Report. Periodic surveillance of pumps and valves is performed in accordance with ASME Code, Section XI, to the extent described in the Inservice Testing Plan, to verify that the systems will provide the flow rates required by the respective analyses. HPCI and RCIC flow tests are performed at two pressures so that the systems' capability to provide rated flow over their operating range is verified. HPSW flow tests verify that rated flow can be delivered to the RHR heat exchangers.

The testing interval for the core and containment cooling systems is based on industry practice, sound engineering judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with frequent tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining redundant cooling systems that the Limiting Conditions for Operation require to be operable during the allowable out-of-service time period. Verifying operability in this context means to administratively ensure that the remaining required systems or subsystems are not known to be inoperable (for example: confirming that equipment necessary for the systems or subsystems to perform their safety functions are not blocked out of service for maintenance). Performance of operability tests is not required.

4.5 I&J Surveillance Requirements Bases

Average and Local LHGR

The LHGR shall be checked daily to determine if fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow and only a few control rods are moved daily, a daily check of power distribution is adequate.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
to ≤ 6.0 mrem to the total body and to ≤ 20.0 mrem to any organ. When the calculated dose from the release of radioactive materials in liquid effluents exceeds any of the above limits, prepare and submit to the Commission within 21 working days, pursuant to Specification 6.9.2, a Special Report which identifies the causes for exceeding the limits and corrective actions that have been taken to reduce the releases of radioactive materials in liquid effluents and proposed corrective actions to be taken to assure that subsequent releases are within the limits. This Special Report shall also include (1) results of radiological analyses of the drinking water source and (2) the radiological impact on the potentially affected drinking water supplies with regard to 40 CFR 141, Safe Drinking Water Act. Reactor shutdown is not required.	
3. During release of radioactive wastes, the following conditions shall be met: a. The minimum dilution water required to satisfy 3.8.B.1 shall be met. b. The gross activity monitor and flow monitor on the waste effluent line shall be operable except as specified in	3a. The liquid radwaste effluents radiation monitor shall be calibrated every 12 months with a known radioactive source positioned in a reproducible geometry with respect to the sensor and every quarter by means of a source check. Additionally, an instrument functional test shall be performed every

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2. The air dose in areas at and beyond the SITE BOUNDARY (see Figure 3.8.1) due to noble gases in gaseous effluents released from the two reactors at the site shall be limited to the following: a. During any calendar quarter for gamma radiation: ≤ 10 mrad. During any calendar quarter for beta radiation: ≤ 20 mrad. b. During any calendar year for gamma radiation: ≤ 20 mrad. During any calendar year for beta radiation: ≤ 40 mrad. When the calculated air dose from radioactive noble gases in gaseous effluents exceeds any of the above limits, prepare and submit to the Commission within 21 working days, pursuant to Specification 6.9.2, a Special Report which identifies the causes for exceeding the limits and defines the corrective actions that have been taken to reduce the releases and proposed corrective actions to be taken to assure that subsequent releases will be within the above limits. Reactor shutdown is not required.	2. Cumulative dose contributions for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per month.
3. The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium and from all radionuclides in particulate form with	3. Cumulative dose contributions for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
half-lives greater than 8 days in gaseous effluents released from the two reactors at the site to areas at and beyond the SITE BOUNDARY (see Figure 3.8.1) shall be limited to the following: a. During any calendar quarter: ≤ 15 mrem. b. During any calendar year: ≤ 30 mrem. When the calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form, with half-lives greater than 8 days in gaseous effluents exceeds any of the above limits, prepare and submit to the Commission within 21 working days, pursuant to Specification 6.9.2, a Special Report. The report shall identify the causes for exceeding the limits and define the corrective actions that have been taken and proposed corrective actions to to assure that subsequent releases will be within the above limits. Reactor shutdown is not required.	shall be determined in accordance with the methodology and parameters in the ODCM at least once per month.
4. During release of gaseous wastes the following conditions shall be met to avoid exceeding the limits specified in 3.8.C.1: a. The main off-gas stack minimum dilution flow of 10,000 cfm shall be maintained. b. One reactor building exhaust vent monitor	4a. The reactor building exhaust vent and main stack noble gas radiation monitors shall be calibrated every 12 months with a known radioactive source positioned in a reproducible geometry with respect to the sensor, and every quarter by means of a functional test. The channel functional test

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
pursuant to Specification 6.9.2 a Special Report which includes the following information: a. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems and the reason for its inoperability. b. Action taken to restore the inoperable equipment to operable status. c. Summary description of action taken to prevent a recurrence. Reactor shutdown is not required.	

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

made, including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the limits have been exceeded. If such is the case, prepare and submit to the Commission, within 21 working days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and schedule for achieving conformance with the above limits. This Special Report shall include an analysis that estimates the radiation exposure to a MEMBER OF THE PUBLIC, including all effluent pathways and direct radiation, including the releases covered by this report, for the calendar year. It shall also describe levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations. If the estimated dose exceeds the above limits and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with 40 CFR 190. Submittal of the report is considered a timely request and a variance is granted until staff action on the request is complete.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Identify the new location in the next Radioactive Dose Assessment Report and include in the report revised figures and tables for the ODCM reflecting the new locations.	
3. Analyses shall be performed on radioactive materials supplied as part of the EPA Environmental Radioactivity Intercomparison Studies Program, or another Interlaboratory Comparison Program that has been approved by the Commission.	3a. A summary of the results obtained as part of the Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.2.
a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence in the Annual Radiological Environmental Operating Report.	
3.8.F Solid Radioactive Waste	4.8.F Solid Radioactive Waste
1. The solid radwaste system shall be used in accordance with a Process Control Program (PCP) to process wet radioactive wastes to meet shipping and burial ground requirements.	1. The PCP shall be used to ensure meeting the burial ground and shipping requirements prior to shipment of radioactive wastes from the site.
a. With the provisions of the Process Control Program not satisfied, suspend shipments of defectively packaged solid radioactive waste from the site. Reactor shutdown is not required.	

LIMITING CONDITIONS FOR OPERATION

3.15 Seismic Monitoring Instrumentation

Applicability

Applies to the operational status of the seismic monitoring instrumentation.

Specifications

A. The seismic monitoring instrumentation shown in Table 3.15 shall be operable.

B. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the NRC pursuant to Specification 6.9.2 within the next 10 working days outlining the cause of the malfunction and the plans for restoring the instrument(s) to operable status.

C. The provisions of Specification 3.0.c are not applicable.

SURVEILLANCE REQUIREMENTS

4.15 Seismic Monitoring Instrumentation

Applicability

Applies to the surveillance requirements of the seismic monitoring instrumentation.

Specifications

A. Each of the required seismic monitoring instruments shall be demonstrated operable by the performance of the Instrument Check, Instrument Functional Test, and Instrument Calibration operations at the frequencies shown in Table 4.15.

B. Each of the required seismic monitoring instruments actuated during a seismic event shall be restored to operable status within 24 hours and an Instrument Calibration performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the NRC pursuant to Specification 6.9.2 within the next 10 working days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

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 Plant Manager or his designated alternate per Specification 6.1.1 prior to implementation and reviewed 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 Plant Manager within 14 days of implementation.

6.8.4 Written procedures shall be established, implemented and maintained covering the activities of the radiological effluent technical specifications as referenced below:

a. Offsite Dose Calculation Manual

b. Quality Assurance Program for the environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.

6.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the NRC in accordance with 10 CFR 50.4, "Written Communications".

6.9.1 Routine Reports (cont'd)

c. Annual Safety/Relief Valve Report

Describe all challenges to the primary coolant system safety and relief valves. Challenges are defined as the automatic opening of the primary coolant safety or relief valves in response to high reactor pressure.

d. 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. Each report shall be submitted no later than the 15th of the month following the calendar month covered by the report.

e. Core Operating Limits Report

(1) Core operating limits shall be established and shall be documented in the CORE OPERATING LIMITS REPORT prior to each Operating Cycle, or prior to any remaining portion of an Operating Cycle, for the following:

a. The APLHGR for Specification 3.5.I,

b. The MCPR for Specification 3.5.K,

c. The K_f core flow adjustment factor for Specification 3.5.K,

d. The LHGR for Specification 3.5.J,

e. The upscale flow biased Rod Block Monitor setpoint and the upscale high flow clamped Rod Block monitor setpoint of Specification 3.2.C.

(2) The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents as amended and approved:

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version)

b. Philadelphia Electric Company Methodologies as described in:

(1) PECo-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code"

6.9.2 Unique Reporting Requirements

Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified herein 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. Report seismic monitoring instrumentation inoperable for more than 30 days (Specification 3.15.B) within the next 10 working days. Submit a seismic event analysis (Specification 4.15.B) within 10 working days of the event.

d. Primary containment leak rate testing approximately three months after the completion of the periodic integrated leak rate test (Type A) required by Specification 4.7.A.2.c.2. For each periodic test, leakage test results from Type A, B and C tests shall be reported. B and C tests are local leak rate tests required by Specification 4.7.A.2.f. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test.

e. Calculated dose from release of radioactive effluents, Specification 3.8.B.2, 3.8.B.4, 3.8.C.2, 3.8.C.3, 3.8.C.5, 3.8.D, and 3.8.E.1.b.

f. Sealed source leakage in excess of limits, Specification 3.13.2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 160 AND 162 TO FACILITY OPERATING

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, UNIT NOS. 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By letter dated January 30, 1990, as supplemented by letter dated April 9, 1991, the Philadelphia Electric Company (PECo or licensee) submitted a request for changes to the Peach Bottom Atomic Power Station, Units 2 and 3, Technical Specifications (TS). The requested changes would revise the testing requirements for other systems or subsystems of the Emergency Core Cooling Systems (ECCS), Reactor Core Isolation Cooling (RCIC) system, and High Pressure Service Water (HPSW) system when one system or subsystem is inoperable; revise the operability requirements of several ECCS and RCIC systems and incorporate some administrative changes. The April 9, 1991, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

Present Peach Bottom Atomic Power Station, Units 2 and 3 Technical Specification surveillance requirements for ECCS, RCIC, and HPSW systems provide for demonstrating the operability of redundant systems or subsystems when one system or subsystem is inoperable. The Emergency Core Cooling Systems are comprised of the Core Spray, Low Pressure Coolant Injection, High Pressure Coolant Injection, and Automatic Depressurization subsystems. These testing requirements are as follows:

(1) One Core Spray subsystem inoperable - demonstrate operability within 24 hours of the operable core spray subsystem and the Low Pressure Coolant Injection (LPCI) subsystems. Demonstrate operability of the same every 72 hours thereafter.

(2) One LPCI pump or subsystem inoperable - demonstrate operability within 24 hours of the operable LPCI pumps/subsystem and core spray subsystems. Demonstrate operability of the same every 72 hours thereafter.

(3) Two HPSW pumps inoperable - demonstrate operability immediately of the operable HPSW pumps. Demonstrate operability of the same weekly thereafter.

(4) Three HPSW pumps inoperable - demonstrate operability immediately of the operable HPSW pump and its diesel generator. Demonstrate operability of the operable HPSW pump weekly thereafter.

(5) One torus cooling loop inoperable - demonstrate operability immediately of the operable torus cooling loop and its diesel generators.

(6) One drywell spray loop inoperable - demonstrate operability immediately of the operable drywell spray loop and its diesel generators.

(7) One torus spray loop inoperable - demonstrate operability immediately of the operable torus spray loop and its diesel generators.

(8) High Pressure Coolant Injection (HPCI) system inoperable - demonstrate operability immediately of the RCIC system and operable subsystems of the ECCS. Demonstrate operability of the RCIC system and ADS actuation logic daily thereafter.

(9) RCIC system inoperable - demonstrate operability immediately of the HPCI subsystem. Demonstrate operability of the HPCI subsystem weekly thereafter.

(10) One Automatic Depressurization System (ADS) valve inoperable - demonstrate operability immediately of the HPCI subsystem and ADS actuation logic for operable ADS valves. Demonstrate operability of the same weekly thereafter.

The licensee proposed to remove the redundant system testing requirements from the ECCS, RCIC, and HPSW systems sections of the Technical Specifications (Section 4.5) while maintaining adequate assurance of system operability needed for accident mitigation.

The requirement for demonstrating operability of the redundant systems identified above for Peach Bottom Atomic Power Station, Units 2 and 3 was originally chosen because there was a lack of plant operating history and a lack of sufficient equipment failure data. Since that time, plant operating experience has demonstrated that testing of the redundant ECCS, RCIC, and HPSW systems when one system or subsystem is inoperable is not necessary to provide adequate assurance of system operability. In fact, taking the redundant system out of service for testing creates the potential of an operator error that would keep the redundant system out of service due to this system being erroneously left in the testing mode. Operability of these systems can be verified by administratively checking equipment status relative to operability requirements.

The current Standard Technical Specifications (STS) and, more specifically, technical specifications approved for recently licensed BWR's accept the philosophy of system operability based on satisfactory performance of monthly, quarterly, operating cycle interval, post maintenance or other

specified performance tests without requiring additional testing when another system is inoperable (except for testing in response to an inoperable diesel generator or inoperable offsite circuit). The staff reviewed PECO's January 30, 1990 submittal to confirm that the testing requirements for the redundant systems or subsystems contained in the existing Technical Specifications, as modified by the proposed amendments, were equivalent with the requirements contained in the Standard Technical Specifications. In Table 1B of PECo's January 30, 1990 submittal, a comparison between the Peach Bottom Technical Specifications and the Standard Technical Specifications was provided.

The staff has reviewed this submittal and determined the proposed Technical Specifications requirements for Peach Bottom are equivalent with those of the Standard Technical Specifications and those of recently licensed BWR's with regard to the testing requirements for redundant systems for the ECCS, RCIC, and HPSW systems.

On this basis, the fact that testing of the redundant system creates the risk of the second system failing, and past operational experience, the staff has determined that the revised testing requirements for the ECCS, RCIC, and HPSW systems are acceptable.

In PECo's safety assessment supporting the subject application and in the proposed 4.5 Bases on page 141, the licensee stated that when one train becomes inoperable (1) the redundant train will be verified to be operable by administratively checking equipment status relative to operability requirements and (2) the nature of and cause for each condition for inoperability should be individually evaluated to identify generic implications, if any, and to determine whether testing of other systems is warranted.

By letter dated January 24, 1991, we requested that the licensee describe on the record what specific actions will be performed to verify operability and how the proposed actions will be implemented. In our letter, we also noted that the proposed wording for the 3.5.A Bases on page 135 could possibly be misconstrued. The licensee responded by letter dated April 9, 1991, expounding on what specific actions would be taken to check equipment status and their process for assessing potential generic implications. Although not requested to do so, the licensee revised the Bases on pages 135 and 141 for both the Unit 2 and 3 TSs to elaborate on what actions would be taken to verify operability and to clarify the Bases for the seven day LCO period when one LPCI subsystem is not available. These two new pages only revised the Bases; there were no changes to the LCOs or Surveillance Requirements in the TSs. The supplemental letter of April 9, 1991 provided clarification but in no way changed the substance of the initial application and did not provide any information that would change or affect the staff's "No Significant Hazards Consideration" as published in the Federal Register on December 26, 1990 (55 FR 53074).

In addition, miscellaneous changes to the Technical Specifications have been proposed which are administrative in nature. The staff has reviewed the proposed changes and determined that the changes achieve consistency throughout

the technical specifications, provide clarifications, or correct errors, and thus are acceptable. The proposed change to the operability requirements for HPCI, RCIC, and ADS to specify that the systems shall be operable whenever there is irradiated fuel in the reactor vessel and reactor steam pressure is greater than 105 psig provides clarification to the current Technical Specifications and is consistent with the function of these systems as described in the Technical Specifications bases and the facility's updated final safety analysis report. On this basis, the proposed change is acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATIONS

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The NRC staff has 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 the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the 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 nor environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: G. Y. Suh, R. Clark

Date: June 12, 1991