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JRBuchanan
RDiggs

Docket Nos. 50-277
and 50-278

OCTOBER 3 1978

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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 46 and 46 to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Units Nos. 2 and 3. The amendments revise the Technical Specifications in response to your requests of May 6, 1977 and March 14, 1978, as supplemented July 11, 1978.

These amendments revise the Technical Specifications to (1) revise the surveillance requirements associated with suppression pool temperature logging, (2) revise the Tables listing Primary Containment Isolation Valves (PCIV) to reflect the addition of a controlled bypass heatup line on the High Pressure Coolant Injection (HPCI) steam supply, and (3) revise the identification of certain valves to reflect plant unique designations between Units 2 and 3.

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

Sincerely,

Original Signed by
T. A. Ippolito
Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 46 to DPR-44
2. Amendment No. 46 to DPR-56
3. Safety Evaluation
4. Notice

cc w/enclosures:
See page 2

Subject to changes noted on p 165 of TS

CCP
Cmsh

OFFICE	ORB #3	ORB #3	OELD	ORB #3	OSE
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DATE	9/13/78	9/18/78	9/29/78	10/3/78	10/3/78

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Education Building
Commonwealth and Walnut Streets
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
License No. DPR-44

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

   A. The applications for amendment by Philadelphia Electric Company, et al, (the licensee) dated May 6, 1977 and March 14, 1978, as supplemented July 11, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 3, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

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

Remove	Insert
165	165
181	181
185	185
186*	186*
187	187
190	190

*No change on this page

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment system.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water volume - 122,900 ft^3
   b. Maximum water volume - 127,300 ft^3

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

1. The suppression chamber water level and temperature shall be checked once per day.

2. a. Whenever there is indication of relief valve operation (except when the reactor is being shutdown and torus cooling is being established) or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

3. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

4. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

TABLE 3.7.1 (Cont'd.)

PRIMARY CONTAINMENT ISOLATION VALVES

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

*Effective upon completion of installation, approved by Amendment No.

TABLE 3.7.4

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

Pen No.			Notes
7A	AO-2-80A; AO-2-86A	MSIV	(1) (2) (3) (5) (9)
7B	AO-2-80B; AO-2-86B	"	"
7C	AO-2-80C; AO-2-86C	"	"
7D	AO-2-80D; AO-2-86D	"	"
8	MO-2-74; MO-2-77		(1) (2) (4) (5) (9)
9A	MO-23-19; MO-23-20; MO-23-21		"
9A	MO-2-38A; MO-2663		"
9B	MO-13-21; MO-13-20; MO-13-30; MO-12-68;		"
9B	MO-2-38B; MO-2663		"
10	MO-13-15; MO-13-16		"
11	MO-23-15; MO-23-16; AO-4807*		"
12	MO-10-17; MO-10-18		"
13A	MO-10-25B; MO-10-154B; SV-4222		"
13B	MO-10-25A; MO-10-154A; SV-4221		"
14	MO-12-15; MO-12-18		"
16A	MO-14-12B; MO-14-11B; SV-4224		"
16B	MO-14-12A; MO-14-11A; SV-4225		"
17	MO-10-32; MO-10-33		"
18	AO-20-82; AO-20-83		"
19	AO-20-94; AO-20-95		"

*Effective upon completion of the modification authorized by Amendment No.

TABLE 3.7.4 (Cont'd.)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

Pen No.		Notes
22	AO-2969A; Check Valve	(1) (2) (4) (5) (10)
25	AO-2520; AO-2505; AO-2519; AO-2521A; AO-2521B	(1) (2) (4) (5) (9)
25	AO-2523; Check Valves	(1) (2) (4) (5)
26	AO-2506; AO-2507	(1) (2) (4) (5) (9)
26	SV-2671G; SV-2978G	(1) (2) (4) (5)
26	AO-2509; AO-2510; AO-4235	(1) (2) (4) (5) (9)
26	SV-4960B; SV-4961B; SV-4966B	(1) (2) (4) (5)
39A	MO-10-31B; MO-10-26B	(1) (2) (4) (5) (9)
39A	SV-4949B; SV-4948B	(1) (2) (4) (5)
39B	MO-10-31A; MO-10-26A	(1) (2) (4) (5) (9)
39B	SV-4949A; SV-4948A	(1) (2) (4) (5)
41	AO-2-39; AO-2-40	(1) (2) (4) (5) (9)
42	Check Valve 11-16, XV-14A, XV-14B	(1) (2) (4) (5) (10)
51A	SV-2671E; SV-2978E	(1) (2) (4) (5)
51B	SV-2671D; SV-2978D	"
51C	SV-2671C; SV-2978C	"
51C	SV-4960C; SV-4961C; SV-4966C	"
51D	SV-2980; Check Valve	"
52F	AO-2969B; Check Valve	(1) (2) (4) (5) (10)
57	AO-2-316; AO-2-317	(1) (2) (4) (5) (9)
203	SV-2671B; SV-2978B	(1) (2) (4) (5)
203	SV-4960D; SV-4961D; SV-4966D	"

TABLE 3.7.4 (Cont'd.)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

Pen No.		Notes
205A	AO-2502B; Check Valve 9-26B	(1) (2) (4) (5) (9)
205B	AO-2502A; Check Valve 9-26A	"
211A	MO-10-38B; MO-10-39B; MO-10-34B	(1) (2) (4) (5) (9)
211A	SV-4951B; SV-4950B	(1) (2) (4) (5)
211B	MO-10-38A; MO-10-39A; MO-10-34A	(1) (2) (4) (5) (9)
211B	SV-4951A; SV-4950A	(1) (2) (4) (5)
212	Check Valve 13-50; AO-4240; AO-4241	(1) (2) (4) (5) (9)
214	Check Valve 23-65; AO-4247; AO-4248	(1) (2) (4) (5) (9)
217B	MO-4244; MO-4244A	(1) (2) (4) (5) (9)
218A	AO-2968	(1) (2) (4) (5) (10)
218B	SV-2671A; SV-2978A	(1) (2) (4) (5)
219	AO-2511; AO-2512; AO-2513; AO-2514	(1) (2) (4) (5) (9)
219	SV-2671F; SV-2978F; SV-4960A SV-4961A; SV-4966A	(1) (2) (4) (5)
221	Check Valve 13-38	(1) (2) (4) (5) (9)
223	Check Valve 23-56	(1) (2) (4) (5) (9)
225	MO-13-41; MO-13-39	(1) (2) (4) (5) (9)
225	MO-14-70; MO-14-71	(1) (2) (4) (5) (9)
227	MO-23-58; MO-23-57	(1) (2) (4) (5) (9)

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

The maximum allowable volume assures the integrity and functional capability of the Suppression Chamber (torus) during postulated LOCA pool swell effects on the torus support system. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in basis 3.5.F.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be repressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of testing which add significant heat, the temperature trends will be closely followed so that appropriate action can be taken if required. Logging is not required during inadvertent relief valve operation since during such periods operator action is actively and directly involved in operations relating to controlling torus temperature and monitoring of temperature trends is a natural part of the operations. Additionally torus temperature is monitored by a recorder during these periods so that an historical record is available.

Operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

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. 46
License No. DPR-56

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

   A. The applications for amendment by Philadelphia Electric Company, et al, (the licensee) dated May 6, 1977 and March 14, 1978, as supplemented July 11, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

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

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: October 3, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

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

Remove	Insert
iv	iv
165	165
181	181
185	185
186	186
187	187
190	190

LIST OF FIGURES

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment system.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water volume - 122,900 ft^3
   b. Maximum water volume - 127,300 ft^3

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

1. The suppression chamber water level and temperature shall be checked once per day.

2. a. Whenever there is indication of relief valve operation (except when the reactor is being shutdown and torus cooling is being established) or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

3. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

4. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

TABLE 3.7.4

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

Pen No.			Notes
7A	AO-2-80A; AO-2-86A	MSIV	(1) (2) (3) (5) (9)
7B	AO-2-80B; AO-2-86B	"	"
7C	AO-2-80C; AO-2-86C	"	"
7D	AO-2-80D; AO-2-86D	"	"
8	MO-2-74; MO-2-77		(1) (2) (4) (5) (9)
9A	MO-23-19; MO-23-20; MO-23-21		"
9A	MO-2-38A; MO-3663		"
9B	MO-13-21; MO-13-20; MO-13-30; MO-12-68;		"
9B	MO-2-38B; MO-3663		"
10	MO-13-15; MO-13-16		"
11	MO-23-15; MO-23-16; AO-5807*		"
12	MO-10-17; MO-10-18		"
13A	MO-10-25B; MO-10-154B; SV-5222		"
13B	MO-10-25A; MO-10-154A; SV-5221		"
14	MO-12-15; MO-12-18		"
16A	MO-14-12B; MO-14-11B; SV-5224		"
16B	MO-14-12A; MO-14-11A; SV-5225		"
17	MO-10-32; MO-10-33		"
18	AO-20-82; AO-20-83		"
19	AO-20-94; AO-20-95		"

*Effective upon completion of the modification authorized by Amendment No.

TABLE 3.7.4 (Cont'd.)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

Pen No.		Notes
22	AO-3969A; Check Valve	(1) (2) (4) (5) (10)
25	AO-3520; AO-3505; AO-3519; AO-3521A; AO-3521B	(1) (2) (4) (5) (9)
25	AO-3523; Check Valves	(1) (2) (4) (5)
26	AO-3506; AO-3507	(1) (2) (4) (5) (9)
26	SV-3671G; SV-3978G	(1) (2) (4) (5)
26	AO-3509; AO-3510; AO-5235	(1) (2) (4) (5) (9)
26	SV-5960B; SV-5961B; SV-5966B	(1) (2) (4) (5)
39A	MO-10-31B; MO-10-26B	(1) (2) (4) (5) (9)
39A	SV-5949B; SV-5948B	(1) (2) (4) (5)
39B	MO-10-31A; MO-10-26A	(1) (2) (4) (5) (9)
39B	SV-5959A; SV-5948A	(1) (2) (4) (5)
41	AO-2-39; AO-2-40	(1) (2) (4) (5) (9)
42	Check Valve 11-16, XV-14A, XV-14B	(1) (2) (4) (5) (10)
51A	SV-3671E; SV-3978E	(1) (2) (4) (5)
51B	SV-3671D; SV-3978D	"
51C	SV-3671C; SV-3978C	"
51C	SV-5960C; SV-5961C; SV-5966C	"
51D	SV-3980; Check Valve	"
52F	AO-3969B; Check Valve	(1) (2) (4) (5) (10)
57	AO-2-316; AO-2-317	(1) (2) (4) (5) (9)
203	SV-3671B; SV-3978B	(1) (2) (4) (5)
203	SV-5960D; SV-5961D; SV-5966D	"

TABLE 3.7.4 (Cont'd.)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

Pen No.		Notes
205A	AO-3502B; Check Valve 9-26B	(1) (2) (4) (5) (9)
205B	AO-3502A; Check Valve 9-26A	"
211A	MO-10-38B; MO-10-39B; MO-10-34B	(1) (2) (4) (5) (9)
211A	SV-5951B; SV-5950B	(1) (2) (4) (5)
211B	MO-10-38A; MO-10-39A; MO-10-34A	(1) (2) (4) (5) (9)
211B	SV-5951A; SV-5950A	(1) (2) (4) (5)
212	Check Valve 13-50; AO-5240; AO-5241	(1) (2) (4) (5) (9)
214	Check Valve 23-65; AO-5247; AO-5248	(1) (2) (4) (5) (9)
217B	MO-5244; MO-5244A	(1) (2) (4) (5) (9)
218A	AO-3968	(1) (2) (4) (5) (10)
218B	SV-3671A; SV-3978A	(1) (2) (4) (5)
219	AO-3511; AO-3512; AO-3513; AO-3514	(1) (2) (4) (5) (9)
219	SV-3671F; SV-3978F; SV-5960A SV-5961A; SV-5966A	(1) (2) (4) (5)
221	Check Valve 13-38	(1) (2) (4) (5) (9)
223	Check Valve 23-56	(1) (2) (4) (5) (9)
225	MO-13-41; MO-13-39	(1) (2) (4) (5) (9)
225	MO-14-70; MO-14-71	(1) (2) (4) (5) (9)
227	MO-23-58; MO-23-57	(1) (2) (4) (5) (9)

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

The maximum allowable volume assures the integrity and functional capability of the Suppression Chamber (torus) during postulated LOCA pool swell effects on the torus support system. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in basis 3.5.F.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be repressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of testing which add significant heat, the temperature trends will be closely followed so that appropriate action can be taken if required. Logging is not required during inadvertent relief valve operation since during such periods operator action is actively and directly involved in operations relating to controlling torus temperature and monitoring of temperature trends is a natural part of the operations. Additionally torus temperature is monitored by a recorder during these periods so that an historical record is available.

Operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 46 AND 46 TO FACILITY LICENSE NOS. DPR-44 AND DPR-56

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

PEACH BOTTOM ATOMIC POWER STATION UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

I. INTRODUCTION

By letters dated May 6, 1977 and March 14, 1978 as supplemented by letter dated July 11, 1978 Philadelphia Electric Company (licensee) proposed changes to the Technical Specifications appended to Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Units Nos. 2 and 3. The changes would (1) revise the surveillance requirements associated with suppression pool temperature logging, (2) revise the Tables listing Primary Containment Isolation Valves (PCIV) to reflect the addition of a controlled bypass heatup line on the High Pressure Coolant Injection (HPCI) steam supply, and (3) revise the identification of certain valves to reflect plant unique designations between Units 2 and 3.

II. EVALUATION

a. Suppression Pool Temperature Logging

The existing Technical Specifications for Peach Bottom Unit Nos. 2 and 3 include the following surveillance:

> "Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated."

The licensee's proposed revision would explicitly delete the logging requirement during periods of inadvertent relief valve operation when the reactor is being shutdown and torus cooling is being established.

The staff's specific requirement associated with this surveillance is to require the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, e.g., surveillance testing of relief valves, RCIC or HPCI operation, such that temperature trends can be closely followed and operator action can be initiated before the temperature reaches the reactor scram setpoint of 110°F. The nature of plant operation during the time that the reactor is being shutdown and torus cooling is being established is that inadvertent operation of a relief valve may occur. During such periods, the operator is actively involved in operations related to the suppression pool, such as placing the RHR system into operation and establishing torus cooling. Operator cognizance of the torus temperature trend is an inherent part of these operations. The staff has determined that since the torus temperature is continually monitored by a recorder, the torus temperature trend data is available to the operator and that the frequent logging of torus temperature serves no unique purpose during periods when the reactor is being shutdown and torus cooling is being established. Therefore, the licensee's request, as proposed, is consistent with staff guidance contained in NUREG-0123. Standard Technical Specifications for General Electric Boiling Water Reactors and is acceptable.

b. HPCI Controlled Bypass Heatup Line Valves

The licensee proposed to install on each of the Units a pressurizing line in parallel with the existing outboard containment isolation valve in the HPCI steam supply line. The purpose of this line is to permit controlled steam line heatup prior to opening the outboard steam line isolation valve and to permit drainage of condensate from between the inboard and outboard isolation valves. Typically, this line will be used when returning the HPCI steam line to service, after it has been removed from service for maintenance during power operation.

The addition of this pressurizing line will not affect the function or intended operation of the HPCI system. Since the addition of this line will circumvent the outboard containment isolation valve on the steam supply line, the licensee has proposed to include a normally closed, air operated isolation valve in the pressurizing line. This valve is designed to fail in a closed position upon a loss of motive power or control power, and will go to a closed position upon the receipt of a HPCI initiation signal. In addition, the valve has been designed to seismic Category I criteria and ASME Section III, Class 2 standards. We have reviewed the licensee's submittal and determined that the proposed modification meets the General Design Criteria for Containment Isolation and the design requirement for an Engineered Safety Feature.

The licensee has also proposed a change to the plant Technical Specifications to incorporate the new containment isolation valve, as well as the existing HPCI steam line exhaust drain valves into the limiting conditions for operation for primary containment isolation valves. We find the proposed modification and associated Technical Specification changes to be acceptable.

c. PCIV Designations

The licensee proposed changes to certain valve designations in Table 3.7.4 "Primary Containment Testable Isolation Valves" to more clearly identify the valves and indicate unique designations for the two Units. We have reviewed the licensee's submittal and determined that the change is pro forma in nature, has no safety or environmental significance and is therefore acceptable.

ENVIRONMENTAL CONSIDERATIONS

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

CONCLUSIONS

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

Dated: October 3, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL

PEACH BOTTOM UNITS NOS. 2 AND 3

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 46 and 46 to Facility Operating License No. DPR-44 and DPR-56, issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company and Atlantic City Electric Company, which revised the Technical Specifications for operation of the Peach Bottom Atomic Power Station Units Nos. 2 and 3, located in York County, Pennsylvania. The amendments are effective as of the date of issuance.

These amendments revise the Technical Specifications to (1) revise the surveillance requirements associated with suppression pool temperature logging, (2) revise the Tables listing Primary Containment Isolation Valves (PCIV) to reflect the addition of a controlled bypass heatup line on the High Pressure Coolant Injection (HPCI) steam supply, and (3) revise the identification of certain valves to reflect plant unique designations between Units 2 and 3.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) applications for amendments dated May 6, 1977 and March 14, 1978, as supplemented July 11, 1978, (2) Amendments Nos. 46 and 46 to License Nos. DPR-44 and DPR-56, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 3rd day of October, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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