REGULATORY DOCKET FILE COPY

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

September 22, 1980

Dockets Nos. 50-277
and 50-278

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

Dear Mr. Bauer:

The Commission has issued the enclosed Amendments Nos. 73 and 72 to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units Nos. 2 and 3. These amendments consist of changes to the Technical Specifications (TSs) in partial response to your application dated May 23, 1980, and complete response to your application dated July 16, 1980.

The changes to the TSs revise the Tables of Containment Isolation Valves and Safety Related Snubbers to reflect modifications you have completed or are planned to satisfy our requirements for: (1) cut and cap of the Control Rod Drive Return Line (CRD RL), (2) Containment Isolation Valve modifications to satisfy NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations", and (3) removal/replacement/addition of safety related snubbers associated with the MK I Containment Long-Term Program. Items (1) and (2) apply to both Peach Bottom 2 & 3. Item (3) applies only to Unit 2.

As stated in our evaluation supporting these amendments, the changes have been approved to conform the TSs with modifications which we have requested. You will be advised separately of our determination on the adequacy of your changes to meet acceptance criteria.

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

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 73 to DPR-44
2. Amendment No. 72 to DPR-56
3. Safety Evaluation
4. Notice

cc w/enclosures: See next page

8010080155

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

cc:

Eugene J. Bradley
Philadelphia Electric Company
Assistant General Counsel
2301 Market Street
Philadelphia, Pennsylvania 19101

Troy B. Conner, Jr.
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

Raymond L. Hovis, Esquire
35 South Duke Street
York, Pennsylvania 17401

Warren K. Rich, Esquire
Assistant Attorney General
Department of Natural Resources
Annapolis, Maryland 21401

Philadelphia Electric Company
ATTN: Mr. W. T. Ullrich
Peach Bottom Atomic
Power Station
Delta, Pennsylvania 17314

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

Curt Cowgill
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Peach Bottom Atomic Power Station
P. O. Box 399
Delta, Pennsylvania 17314

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
US EPA
Crystal Mall #2
Arlington, Virginia 20460

Region III Office
ATTN: EIS COORDINATOR
Curtis Building (Sixth Floor)
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

M. J. Cooney, Superintendent
Generation Division - Nuclear
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Government Publications Section
State Library of Pennsylvania
Education Building
Commonwealth and Walnut Streets
Harrisburg, Pennsylvania 17126

cc w/enclosure(s) & incoming dtd.:
5/23 & 7/16/80
Mr. R. A. Heiss, Coordinator
Pennsylvania State Clearinghouse
Governor's Office of State Planning
and Development
P. O. Box 1323
Harrisburg, Pennsylvania 17120

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. 73
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 23 and July 16, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 73, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

[redacted]

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 22, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 73

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 Page | Insert Page |
|---|---|
| 180 | 180 |
| 186 | 186 |
| 187 | 187 |
| 234f | 234f |
| 234n | 234n |
| - | 234o (new page) |
| 257a | 257a |

TABLE 3.7.1 (Cont'd.)

PRIMARY CONTAINMENT ISOLATION VALVES

| Group | Valve Identification | Number of Power Operated Valves | | Maximum Operating Time (sec.) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| | | Inboard | Outboard | | | |
| NA | Feedwater check valves | 2* | 2* | NA | O | Process |
| 3 | Radioactive gas sample isolation valves | 4 | 1# | NA | C | SC |
| 3 | Instrument nitrogen compressor suction line isolation valves | 1# | 1 | 5 | O | GC |
| 3 | Oxygen Analyzer System | | 14 | NA | O | GC |
| NA | Standby liquid control system check valves | 1* | 1* | NA | C | Process |
| 2B | RHRS shutdown cooling suction isolation valves | 1 | 1 | 32 | C | SC |
| 2B | RHRS shutdown cooling injection isolation valves | | 1 | 24 | C | SC |
| 2B | RHRS Reactor Vessel Head Spray isolation valves | 1 | 1 | 30 | C | SC |
| 2C | Feedwater Flush Valves | | 2 | 50 | C | SC |
| 4 | HPCIS steam line isolation valves | 1 | 1 | 20 | O | GC |
| 5 | RCICS steam line isolation valves | 1 | 1 | 15 | O | GC |
| 2A | Reactor water cleanup system isolation valves | 1 | 1 | 30 | O | GC |
| 2A | Reactor water cleanup system return isolation valve | | 1 | 20 | O | GC |

*valves not power operated.
#Effective upon completion of the modification required by NUREG 0578, item 2.1.5a

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) |
| 26 | SV-8100* | (1)(2)(4)(5) |
| 26, 51C 203, 219 | SV-8101* | (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) |
| 102B | Breathing Air System-2 Gate Valves* | (1)(2)(4)(5)(9) |
| 203 | SV-2671B; SV-2978B | (1)(2)(4)(5) |

TABLE 3.7.4 (Cont'd.)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

| Pen No. | | Notes |
|---|---|---|
| 203 | SV-4960D; SV-4961D; SV-4966D | (1) (2) (4) (5) |
| 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) |

* Effective upon completion of the modification

TABLE 3.11.D.1

Safety Related Shock Suppressors (Snubbers)

| SNUBBER NUMBER | LOCATION | ELEVATION | SNUBBER IN HIGH (1) RADIATION AREA DURING SHUTDOWN | SNUBBERS ESPECIALLY DIFFICULT TO REMOVE | SNUBBERS INACCESSIBLE DURING NORMAL OPERATION | SNUBBERS ACCESSIBLE DURING NORMAL OPERATION |
|---|---|---|---|---|---|---|
| | | | | Unit 2 (cont'd) | | |
| 1-GG-S-35 | MSRV | 155 | See 4.11.D.4.b | X | Drywell | |
| 1-GG-S-36 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-63 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-64 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-65 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-66 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-67 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-68 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-69 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-72 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-74 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-75 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-76 | MSRV | 155 | " | X | Drywell | |
| 1-GG-S-77 | MSRV | 155 | " | X | Drywell | |

TABLE 3.11.D.1

Safety Related Shock Suppressors (Snubbers)

| SNUBBER NUMBER | LOCATION | ELEVATION | SNUBBER IN HIGH (1) RADIATION AREA DURING SHUTDOWN | SNUBBERS ESPECIALLY DIFFICULT TO REMOVE | SNUBBERS INACCESSIBLE DURING NORMAL OPERATION | SNUBBERS ACCESSIBLE DURING NORMAL OPERATION |
|---|---|---|---|---|---|---|
| 23-DBN-S-28 | HPCI | 117 | See 4.11.D.4.b | X | | Torus Room |
| 23-DBN-S-29 | HPCI | 117 | " | X | | Torus Room |
| 23-HB-S-30 | HPCI | 93 | " | | | HPCI Room |
| 23-HB-S-36 | HPCI | 103 | " | | | HPCI Room |
| 23-HB-S-37 | HPCI | 103 | " | | | HPCI Room |
| 23-HB-S-38 | HPCI | 126 | " | X | | Torus Room |
| 1-GG-S-101-A | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-101-B | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-102-A | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-102-B | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-103-A | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-103-B | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-104-A | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-104-B | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-105-A | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-105-B | MSRV | 135 | " | X | Drywell | |
| 1-GG-S-106-B | MSRV | 135 | " | X | Drywell | |

TABLE 3.11.D.1

Safety Related Shock Suppressors (Snubbers)

| SNUBBER NUMBER | LOCATION | ELEVATION | SNUBBER IN HIGH (1) RADIATION AREA DURING SHUTDOWN | SNUBBERS ESPECIALLY DIFFICULT TO REMOVE | SNUBBERS INACCESSIBLE DURING NORMAL OPERATION | SNUBBERS ACCESSIBLE DURING NORMAL OPERATION |
|---|---|---|---|---|---|---|
| 1-GG-S-106-B | MSRV | 135 | See 4.11.D.4.b | X | Drywell | |
| 1-GG-S-108-A | MSRV | 145 | " | X | Drywell | |
| 1-GG-S-108-B | MSRV | 145 | " | X | Drywell | |
| 1-GG-S-109-A | MSRV | 145 | " | X | Drywell | |
| 1-GG-S-109-B | MSRV | 145 | " | X | Drywell | |
| 1-GG-S-110-A | MSRV | 145 | " | X | Drywell | |
| 1-GG-S-110-B | MSRV | 145 | " | X | Drywell | |
| 1-GG-S-111-A | MSRV | 145 | " | X | Drywell | |
| 1-GG-S-111-B | MSRV | 145 | " | X | Drywell | |
| 1-GG-S-112-A | MSRV | 145 | " | X | Drywell | |
| 1-GG-S-112-B | MSRV | 145 | " | X | Drywell | |

Notes for Table 3.11.D.1

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

6.9.3 Continued

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

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

g. Effluent Releases

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

1) Gaseous Releases

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

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

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. 72
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 23 and July 16, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 72, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 22, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 72

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 Page | Insert Page |
|---|---|
| 180 | 180 |
| 186 | 186 |
| 187 | 187 |
| 257a | 257a |

TABLE 3.7.1 (Cont'd.)

PRIMARY CONTAINMENT ISOLATION VALVES

| Group | Valve Identification | Number of Power Operated Valves | | Maximum Operating Time (sec.) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| | | Inboard | Outboard | | | |
| NA | Feedwater check valves | 2* | 2* | NA | O | Process |
| 3 | Radioactive gas sample isolation valves | 4 | 1# | NA | C | SC |
| 3 | Instrument nitrogen compressor suction line isolation valves | 1# | 1 | 5 | O | GC |
| 3 | Oxygen Analyzer System | | 14 | NA | O | GC |
| NA | Standby liquid control system check valves | 1* | 1* | NA | C | Process |
| 2B | RHRS shutdown cooling suction isolation valves | 1 | 1 | 32 | C | SC |
| 2B | RHRS shutdown cooling injection isolation valves | | 1 | 24 | C | SC |
| 2B | RHRS Reactor Vessel Head Spray isolation valves | 1 | 1 | 30 | C | SC |
| 2C | Feedwater Flush Valves | | 2 | 50 | C | SC |
| 4 | HPCIS steam line isolation valves | 1 | 1 | 20 | O | GC |
| 5 | RCICS steam line isolation valves | 1 | 1 | 15 | O | GC |
| 2A | Reactor water cleanup system isolation valves | 1 | 1 | 30 | O | GC |
| 2A | Reactor water cleanup system return isolation valve | | 1 | 20 | O | GC |

*valves not power operated.
#Effective upon completion of the modification required by NUREG 0578, item 2.1.5a

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) |
| 26 | SV-9100* | " |
| 26, 51C 203, 219 | SV-9101* | " |
| 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) |
| 102B | Breathing Air System-2 Gate Valves* | " |
| 203 | SV-3671B; SV-3978B | (1)(2)(4)(5) |

TABLE 3.7.4 (Cont'd.)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

| Pen No. | | Notes |
|---|---|---|
| 203 | SV-5960D; SV-5961D; SV-5966D | (1) (2) (4) (5) |
| 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) |

* Effective upon completion of the modification.

6.9.3 Continued

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

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

g. Effluent Releases

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

1) Gaseous Releases

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 73 TO LICENSE NO. DPR-44
AND
AMENDMENT NO. 72 TO LICENSE NO. DPR-56

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

PHILADELPHIA ELECTRIC COMPANY

DOCKETS NOS. 50-277 AND 50-278

## Introduction

By letters dated May 23[1] and July 16[2], 1980 Philadelphia Electric Company (licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units Nos. 2 and 3. Included in these requests were proposed changes to the Tables of Containment Isolation Valves and Safety Related Snubbers to reflect modifications completed or planned by the licensee to satisfy our requirements for: (1) cut and cap of the Control Rod Drive Return Line (CRD RL), (2) Containment Isolation Valve modifications to satisfy NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations", and (3) removal/replacement/addition of safety related snubbers associated with the MK I Containment Long-Term Program. Items (1) and (2) apply to both Peach Bottom 2 and 3. Item (3) applies only to Unit 2.

## Evaluation

### 1. CRD RL Modification

The generic problem of CRD RL nozzle cracking is described in NUREG-0312[3]. Our interim criteria for resolution of this issue, as stated in NUREG-0312 involved cutting and capping of the CRD RL and rerouting of the system flow to either the Reactor Water Cleanup System or feedwater lines. In response to this initial position, the licensee indicated his intentions[4] to reroute the CRD RL fluid. This action was reviewed by the NRC staff and incorporated into the TSs by Amendments Nos. 60 & 60 to DPR-44 and DPR-56[5]. Subsequent to this action, the NRC staff issued its final position regarding CRD RL nozzle cracking which provided acceptance criteria for isolating the CRD RL without rerouting[6,7]. Our letter on this matter authorized such action by licensees contingent upon satisfying certain acceptance criteria. The licensee's request dated July 16, 1980, stated that he has satisfied the requirements for cut and cap of the CRD RL without rerouting in accordance with our requirements as applicable to the Peach Bottom Units. Accordingly, we find that the licensee's submittal is an administrative change to conform the TSs to planned and implemented modifications with post-implementation review by the NRC staff. This change is thus approved for the purpose of verification by our Office of Inspection and Enforcement (OIE) of licensee actions. Verification of the adequacy of licensee actions to satisfy our acceptance criteria for CRD RL cut and cap without rerouting will be performed

by OIE. Results of this verification will be reported in routine Inspection Reports.

2. Containment Isolation Valve Modifications

The licensee has proposed to install additional containment isolation valves in the radioactive gas sampler line and in the instrument nitrogen compressor suction line to meet the NRC staff's requirements of Item 2.1.5.a of NUREG-0578 (8,9). We have previously reviewed and evaluated this proposed action by the licensee(10). Accordingly, the proposed change to the TSs is considered an administrative action to conform the facility design to our stated requirements and is acceptable.

The licensee's submittal of May 23, 1980, also proposed the addition of containment isolation valves for a permanent breathing air header system for contaminated areas inside containment. These lines will be equipped with locked closed valves and will be provided with test connections for leak rate testing.

We have reviewed the licensee's proposed modification and determined that it satisfies our acceptance criteria for containment isolation valves and is acceptable. The design is consistent with our recommendations set forth in NUREG-0578.

3. Safety Related Snubbers

By our letter dated October 31, 1979 (11), we provided to the licensee our acceptance criteria for the MK I Containment Long-Term Program. Licensees of all facilities with MK I type containments are to effect necessary modifications to satisfy these requirements and to provide a plant-unique analysis to demonstrate conformance.

In response to this requirement, the licensee installed, during the current (Summer 1980) refueling outage of Unit No. 2, vacuum breakers on the main steam safety relief valve discharge piping. This modification necessarily affected the safety related snubbers associated with this piping. Therefore, the licensee's request of May 23, 1980, proposed changes to the table of safety related snubbers to reflect the NRC staff required modifications.

We have reviewed the licensee's request and determined that the proposed change is administrative in nature for the purpose of verification by OIE of licensee actions. Our acceptance of the licensee's actions to satisfy the MK I Long-Term Program requirements (including the adequacy and identification of required safety related snubbers) will be performed as part of our evaluation of the MK I Containment Long-Term Program evaluation.

In view of the above, we find the licensee's request to be pro forma in nature and is acceptable.

4. Miscellaneous Changes

The only other change to the Peach Bottom Units Nos. 2 and 3 specifications authorized by these amendments is correction of a typographical error associated with sealed source leakage reporting requirements. Our review of the proposed change indicates that the revision is a correction and is 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 Section 51.5(d)(4) that an environmental impact statement or negative declaration and environmenal 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:** September 22, 1980

## References

1. Letter, PECO (Bradley) to NRC (Denton), May 23, 1980.

2. Letter, PECO (Bradley) to NRC (Denton), July 16, 1980.

3. NUREG-0312 "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Cracking", July 1977.

4. Letter, PECO (Bradley) to NRC (Denton), August 27, 1980.

5. Letter, NRC (Ippolito) to PECO (Bauer), September 19, 1979.

6. Letters, NRC (Ippolito) to PECO (Bauer), April 7 and 23, 1980.

7. NUREG-0619 (for comment) "BWR Feedwater Nozzle and Control Rod Return Line Nozzle Cracking", April 1980.

8. NUREG-0578 "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations", July 1979.

9. Letter, NRC (Eisenhut) to ALL OPERATING NUCLEAR POWER PLANTS, September 13, 1979.

10. Letter, NRC (Ippolito) to PECO (Bauer), February 28, 1980.

11. Letter, NRC (Eisenhut) to PECO (Bauer), October 31, 1979.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 73 and 72 to Facility Operating Licenses Nos. 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 Technical Specifications for operation of the Peach Bottom Atomic Power Station, Units Nos. 2 and 3 (the facility) located in York County, Pennsylvania. The amendments are effective as of the date of issuance.

The amendments to the Technical Specifications revise the Tables of Containment Isolation Valves and Safety Related Snubbers to reflect modifications completed or planned by the licensee to satisfy the Commission's requirements for: (1) cut and cap of the Control Rod Drive Return Line, (2) Containment Isolation Valve modifications to satisfy NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations", and (3) removal/replacement/addition of safety related snubbers associated with the MK I Containment Long-Term Program. Items (1) and (2) apply to both Peach Bottom 2 and 3. Item (3) applies only to Unit 2.

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) the applications for amendments dated May 23 and July 16, 1980, (2) Amendments Nos. 73 and 72 to Licenses Nos. DPR-44 and DPR-56, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW, Washington, DC, and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 22nd day of Septemember 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing