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

July 2, 1984

Dockets Nos. 50-277
and 50-278

Posted
Amdt. 102
to DPR-56

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

Dear Mr. Bauer:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENTS PERTAINING TO TMI ACTION PLANS SPECIFIED IN GENERIC LETTER 83-02

The Commission has issued the enclosed Amendments Nos. 100 and 102, to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units Nos. 2 and 3. These amendments consist of Technical Specifications (TSs) changes to add operability and surveillance requirements addressing TMI Action Plan items II.K.3.3 (Reporting of Safety and Relief Valve Failures and Challenges), II.K.3.13 (High-Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation), and II.K.3.15 (Isolation of High-Pressure Coolant Injection and Reactor Core Isolation Cooling Modifications). These TS change requests were proposed in your letter dated February 11, 1982, as supplemented by a letter dated August 24, 1983. This last submittal was in response to our guidance provided in Generic Letter (G.L.) 83-02.

These amendments will require the reporting of safety/relief valve failures and challenges (II.K.3.3), will change the TSs for reactor core isolation cooling (RCIC) restart (II.K.3.13) and will provide surveillance requirements on the time delay relay included in the high pressure coolant injection (HPCI) and RCIC systems in accordance with the guidance provided in G.L. 83-02.

The licensee has also proposed TSs to Limit Overtime (I.A.1.3) and RCIC Suction II.K.3.22). Our review of these proposed TSs changes is still in progress. Our staff is currently in contact with your staff concerning these two items. As a result of these discussions, revisions to your proposed TS requirements for I.A.1.3 and II.K.3.22 may be required in the near future. Resolution of these issues together with future II.E.4.2.7 TS requirements will be provided under new plant specific tasks.

Based upon the above, we conclude that Multiple Plant Action B-72 should now be considered complete.

A copy of the Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's next monthly Federal Register. In addition, we are including for your information a Peach Bottom status summary of NUREG-0737 Technical Specification Requirements pertaining to Generic Letter 83-02.

If you have any questions, please contact me.

Sincerely,

Gerald E. Gears, Project Manager
Operating Reactors Branch No. 4
Division of Licensing

Enclosures:
1. Status Summary
2. Amendment No. 100 to DPR-44
3. Amendment No. 102 to DPR-56
4. Safety Evaluation

Philadelphia Electric Company

cc w/enclosure(s):

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STATUS SUMMARY OF NUREG-0737 TECHNICAL SPECIFICATIONS

(GENERIC LETTER 83-02) PEACH BOTTOM UNITS 2 AND 3

1. STA Training (I.A.1.1.3)

   No action is required by the licensee pending decision regarding engineering expertise on shift by the Commission.

2. Limit Overtime (I.A.1.3)

   The licensee has submitted proposed changes in the Technical Specifications (TSs) in response to Generic Letter 83-02. We reviewed these TSs and identified some areas where the licensee needed to revise the TSs. The revised TSs will be submitted by the licensee in the near future. This is to be handled as a plant specific task.

3. Dedicated Hydrogen Penetration (II.E.4.1)

   Technical Specifications required by this Action Plan item were approved on September 30, 1980, as Amendments Nos. 73 and 72 to the Peach Bottom Operating Licenses for Units 2 and 3, respectively. This item is complete.

4. Containment Pressure Setpoint (II.E.4.2.5)

   By a letter dated December 15, 1981, the staff informed the licensee that no changes in the containment pressure setpoint are required for Peach Bottom Units 2 and 3. This item is complete.

5. Containment Purge Valves (II.E.4.2.6)

   This is a plant specific item. The Technical Specifications for this item will be reviewed and approved as part of the MPA B-24 review.

6. Radiation Signal on Purge Valves (II.E.4.2.7)

   On May 31, 1983, the staff issued a Safety Evaluation Report for the submittal generated by the BWR Owner's Group for this item. The licensee will be required to submit Technical Specifications after hardware modifications are completed. This is to be handled as a plant specific task.

7. Reporting SV and RV Failures and Challenges (II.K.3.3)

   The licensee has submitted proposed changes in the Technical Specifications to implement the requirements of this item. Our Safety Evaluation is enclosed. This item is complete.

8. RCIC Restart and RCIC Suction (II.K.3.13, II.K.3.22)

The licensee has proposed changes in the TSs to include the requirements of these two items in the TSs for both Peach Bottom Units. Our Safety Evaluation for RCIC Restart (II.K.3.13) is enclosed. We find TSs for this item to be acceptable.

We have also reviewed the proposed TSs for RCIC suction transfer (II.K.3.22). We have determined that the licensee has not included adequate surveillance requirements for testing RCIC suction transfer capability. The licensee should verify (once/operating cycle) that the suction of the RCIC is automatically transfered from the condensate storage tank to the suppression pool on a condensate storage tank water low-level signal. However, the proposed TSs for instrumentation which initiate the RCIC suction transfer are acceptable. This is to be handled as a plant specific task.

9. Isolation of HPCI and RCIC Modification (II.K.3.15)

The licensee has submitted proposed changes in the TSs by a letter dated February 11, 1982. Our Safety Evaluation is enclosed. This item is complete.

10. Interlock on Recirculation Pump Loops (II.K.3.19)

This item is not applicable to Peach Bottom Units 2 and 3.

11. Common Reference Level (II.K.3.27)

No changes in Technical Specifications for Peach Bottom Units 2 and 3 are required to reflect the common reference level established by this item. This item is complete.

12. Manual Depressurization (II.K.3.45)

No changes in the Technical Specifications are required by this item.

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. 100
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 February 11, 1982, as supplemented by letter dated August 24, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 100, 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

George W Rivenbark

George W. Rivenbark, Chief
Operating Reactors Branch No. 4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 2, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 100

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

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

Remove	Insert
70	70
81	81
81a	81a
130	130
254a	254a
254b	254b
255	255
256	256
	256a

TABLE 3.2.B (CONTINUED)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	RCIC Turbine High Flow	$\leq$450" H_2O(2)	2 Inst. Channels	
1	RCIC Turbine High Flow Time Delay	3 ± 1 seconds	2 Inst. Channels	
2	RCIC Turbine Compartment Wall	$\leq$200 deg. F (2)	4 Inst.] 16 Inst.	
6	RCIC Steam Line Area Temp.	$\leq$200 deg. F (2)	12 Inst.] 16 Inst.	
2	RCIC Steam Line Low Pressure	100> p> 50 psig (2)	4 Inst.	
1	HPCI Turbine Steam Line High Flow	$\leq$225" H_2O (3)	2 Inst. Channels	
1	HPCI Turbine High Flow Time Delay	3 ± 1 seconds	2 Inst. Channels	

TABLE 4.

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor Water Level (7)	(1) (3)	Once/operating cycle	Once/day
2) Drywell Pressure (7)	(1) (3)	Once/operating cycle	Once/day
3) Reactor Pressure (7)	(1) (3)	Once/operating cycle	Once/day
4) Auto Sequencing Timers	NA	Once/operating cycle	None
5) ADS - LPCI or CS Pump Disch. Pressure Interlocks	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	NA	None
7) Core Spray Sparger d/p	(1)	Once/6 months	Once/day
8) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
9) Steam Line High Flow Timers (HPCI and RCIC)	NA	Once/operating cycle	None
10) Steam Line High Temp. (HPCI & RCIC)	(1) (3)	Once/operating cycle	Once/day
11) Safeguards Area High Temp.	(1)	Once/3 months	None
12) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None

Amendment No. ~~88~~, ~~97~~, 100

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

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.D. Reactor Core Isolation Cooling (RCIC Sub-System)	4.5.D. Reactor Core Isolation Cooling (RCIC Sub-System)

LIMITING CONDITIONS FOR OPERATION

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

	Item	Frequency
(a)	Simulated Automatic Actuation Test*	Once/Operating Cycle
(b)	Pump Operability	Once/Month
(c)	Motor Operated Valve Operability	Once/Month
(d)	Flow Rate at ~ 1000 psig Steam Pressure	Once/3 Months
(e)	Flow Rate at ~ 150 psig Steam Pressure	Once/Operating Cycle

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

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

*Shall include automatic restart on low water level signal.

6.9.1.a (Continued)

Start-up reports shall be submitted within 90 days following resumption or commencement of commercial full power operation.

b. Annual Occupational Exposure Tabulation (1)

A tabulation shall be made on an annual basis of the number of station utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job function, (2) e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation shall be submitted for the previous calendar year prior to March 1 of each year. The dose assignment to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total wholebody dose received from external sources shall be assigned to specific major job functions.

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 to the Office of Management and Program Analysis (or its successor), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy of the appropriate Regional Office, to be submitted no later than the 15th of the month following the calendar month covered by the report.

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(1) A single submittal may be made for a multiple unit station.

(2) This tabulation supplements the requirements of 10 CFR 20.407.

6.9.2 Reportable Occurrences

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

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

(1) Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 2.a(5), 2.a(6), or 2.b(1) below.

(2) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

6.9.2 (Continued)

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 2.b(2) below.

(3) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

Note: Leakage of valve packing or gaskets within the limits for identified leagage set forth in technical specifications need not be reported under this item.

(4) Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions greater than or equal to 1.0% delta k/k; calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% delta k/k; or occurrence of any unplanned criticality.

(5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

(6) Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

6.9.2 (Continued)

Note: For items 2.a(5) and 2.a(6) reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 2.b(2) and 2.b(3) below.

(7) Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

(8) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

(10) Failure of a primary coolant system safety or relief valve to close.

6.9.2 (Continued)

b. Thirty-Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minumum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 2.b(1) and 2.b(2) need not be reported except where test results themselves reveal a degraded mode as described above.

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.102
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 February 11, 1982, as supplemented by letter dated August 24, 1983, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

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

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.102 , 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

George W. Rivenbark

George W. Rivenbark, Acting Chief
Operating Reactors Branch No. 4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 2, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

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

Remove	Insert
70	70
81	81
81a	81a
130	130
254a	254a
254b	254b
255	255
256	256
	256a

TABLE 3.2.B (CONTINUED)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Of Operable Instrument Channels Per Trip System(1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	RCIC Turbine High Flow	$\leq$450" H_2O(2)	2 Inst. Channels	
1	RCIC Turbine High Flow Time Delay	3 ± 1 seconds	2 Inst. Channels	
2	RCIC Turbine Compartment Wall	$\leq$200 deg. F (2)	4 Inst.]]16 Inst.	
6	RCIC Steam Line Area Temp.	$\leq$200 deg. F (2)	12 Inst.]	
2	RCIC Steam Line Low Pressure	100> p> 50 psig (2)	4 Inst.	
1	HPCI Turbine Steam Line High Flow	$\leq$225" H_2O (3)	2 Inst. Channels	
1	HPCI Turbine High Flow Time Delay	3 ± 1 seconds	2 Inst. Channels	

Amendment No. 74, 102

TABLE 4
MINIMUM TEST AND CAL[illegible]TION FREQUENCY FOR CSCS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor Water Level (7)	(1) (3)	Once/operating cycle	Once/day
2) Drywell Pressure (7)	(1) (3)	Once/operating cycle	Once/day
3) Reactor Pressure (7)	(1) (3)	Once/operating cycle	Once/day
4) Auto Sequencing Timers	NA	Once/operating cycle	None
5) ADS - LPCI or CS Pump Disch. Pressure Interlocks	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	NA	None
7) Core Spray Sparger d/p	(1)	Once/6 months	Once/day
8) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
9) Steam Line High Flow Timers (HPCI and RCIC)	NA	Once/operating cycle	None
10) Steam Line High Temp. (HPCI & RCIC)	(1) (3)	Once/operating cycle	Once/day
11) Safeguards Area High Temp.	(1)	Once/3 months	None
12) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None

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

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

Amendment No. 97, 102

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

	Item	Frequency
(a)	Simulated Automatic Actuation Test*	Once/Operating Cycle
(b)	Pump Operability	Once/Month
(c)	Motor Operated Valve Operability	Once/Month
(d)	Flow Rate at ~ 1000 psig Steam Pressure	Once/3 Months
(e)	Flow Rate at ~ 150 psig Steam Pressure	Once/Operating Cycle

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

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

*Shall include automatic restart on low water level signal.

6.9.1.a (Continued)

Start-up reports shall be submitted within 90 days following resumption or commencement of commercial full power operation.

b. Annual Occupational Exposure Tabulation (1)

A tabulation shall be made on an annual basis of the number of station utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job function, (2) e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation shall be submitted for the previous calendar year prior to March 1 of each year. The dose assignment to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total wholebody dose received from external sources shall be assigned to specific major job functions.

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 to the Office of Management and Program Analysis (or its successor), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy of the appropriate Regional Office, to be submitted no later than the 15th of the month following the calendar month covered by the report.

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(1) A single submittal may be made for a multiple unit station.

(2) This tabulation supplements the requirements of 10 CFR 20.407.

6.9.2 Reportable Occurrences

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

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

(1) Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 2.a(5), 2.a(6), or 2.b(1) below.

(2) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

6.9.2 (Continued)

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 2.b(2) below.

(3) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

Note: Leakage of valve packing or gaskets within the limits for identified leagage set forth in technical specifications need not be reported under this item.

(4) Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions greater than or equal to 1.0% delta k/k; calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% delta k/k; or occurrence of any unplanned criticality.

(5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

(6) Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

6.9.2 (Continued)

Note: For items 2.a(5) and 2.a(6) reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 2.b(2) and 2.b(3) below.

(7) Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

(8) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

(10) Failure of a primary coolant system safety or relief valve to close.

6.9.2 (Continued)

b. Thirty-Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minumum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 2.b(1) and 2.b(2) need not be reported except where test results themselves reveal a degraded mode as described above.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 100 AND 102 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, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1. INTRODUCTION

In November 1980, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements" which included all TMI Action Plan items approved by the Commission for implementation at nuclear power reactors. NUREG-0737 identifies those items for which Technical Specifications are required. A number of items which require Technical Specifications were scheduled for implementation by December 31, 1981. The staff provided guidance on the scope of Technical Specifications for all of these items in Generic Letter 83-02. Generic Letter 83-02 was issued to all Boiling Water Reactor (BWR) licensees on January 10, 1983. In Generic Letter 83-02 the staff requested licensees to:

a. review their facility's Technical Specifications to determine if they were consistent with the guidance provided in the generic letter, and

b. submit an application for a license amendment where deviations or absence of Technical Specifications were found.

By letter dated February 11, 1982, Philadelphia Electric Company (licensee) submitted a Technical Specification change request for Peach Bottom Units 2 and 3 which included many items covered by Generic Letter 83-02. By letter dated August 24, 1983, the licensee responded to Generic Letter 83-02 and submitted a Technical Specification change request for Peach Bottom Units 2 and 3. The later submittals revised some of the Technical Specifications proposed in the February 11, 1982 letter. This evaluation covers the following TMI Action Plan items:

a. Reporting of Safety and Relief Valve Failures and Challenges (II.K.3.3)

b. RCIC Restart (II.K.3.13)

c. Isolation of HPCI and RCIC Modification (II.K.3.15)

2. DISCUSSION AND EVALUATION

A. Reporting of Safety/Relief Valve Failures and Challenges (II.K.3.3)

In Generic Letter 83-02, the staff requested licensees to formalize the reporting requirements for safety/relief valve failures and challenges. The licensee has proposed Technical Specifications which will require the licensee to report the failures promptly with written follow-up, and the challenges in an annual report. The proposed Technical Specifications meet the intent of the guidance provided in Generic Letter 83-02. Therefore we find it acceptable.

B. RCIC Restart (II.K.3.13)

TMI Action Plan Item II.K.3.13 recommends modifications to the Reactor Core Isolation Cooling System (RCIC) such that:

1. The system will restart on a subsequent low water level signal after it has been terminated by a high water level in the reactor vessel.

In Generic Letter 83-02, the staff provided guidance on necessary changes in the Technical Specifications for implementation of the modifications. The proposed changes in Technical Specifications for RCIC are in response to Generic Letter 83-02. We have reviewed the proposed changes in the Technical Specifications and determined that the changes are consistent with the guidance provided in Generic Letter 83-02. We find the changes acceptable.

C. Isolation of HPCI and RCIC Modifications (II.K.3.15)

TMI Action Plan Item II.K.3.15 recommends that the pipe-break-detection circuitry should be modified so that pressure spikes resulting from High-Pressure Coolant Injection (HPCI) and RCIC system initiation will not cause inadvertent system isolation. The licensee has completed the modification recommended by this item.

In Generic Letter 83-02, the staff provided guidance on the scope of the Technical Specifications required by this item. The licensee has proposed changes in the Technical Specifications for Peach Bottom Units 2 and 3. We have reviewed the proposed changes for both Units 2 and 3 and determined that the Technical Specifications cover the surveillance requirements on the time delay relay included in HPCI and RCIC systems. The proposed changes are consistent with our gudiance in Generic Letter 83-02. Therefore, we find the proposed changes to be acceptable.

3. ENVIRONMENTAL CONSIDERATION

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

4. CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 2, 1984

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