March 3, 1988

Dockets Nos. 50-277/278

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

Dear Mr. Bauer:

SUBJECT: HYDROGEN WATER CHEMISTRY (TAC NOS. 64809, 64810)

RE: PEACH BOTTOM ATOMIC POWER STATION, UNIT NOS. 2 AND 3

The Commission has issued the enclosed Amendments Nos. 129 and 132 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3. These amendments consist of changes to the Technical Specifications in response to your application dated February 12, 1987 as supplemented on October 20, 1987.

This amendment would revise the Technical Specifications with changes related to implementation of a Hydrogen Water Chemistry (HWC) program to improve reactor water chemistry and thus to reduce the potential for intergranular stress corrosion cracking.

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

Sincerely,

/s/

Robert E. Martin, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 129 to DPR-44
2. Amendment No. 132 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page

8803210268 880303
PDR ADOCK 05000277
P PDR

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RMartin:mr	SETurk	WButler
2/24/88	01/05/88	2/25/88	3/3/87

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

March 3, 1988

Dockets Nos. 50-277/278

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

Dear Mr. Bauer:

SUBJECT: HYDROGEN WATER CHEMISTRY (TAC NOS. 64809, 64810)

RE: PEACH BOTTOM ATOMIC POWER STATION, UNIT NOS. 2 AND 3

The Commission has issued the enclosed Amendments Nos. 129 and 132 to Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3. These amendments consist of changes to the Technical Specifications in response to your application dated February 12, 1987 as supplemented on October 20, 1987.

This amendment would revise the Technical Specifications with changes related to implementation of a Hydrogen Water Chemistry (HWC) program to improve reactor water chemistry and thus to reduce the potential for intergranular stress corrosion cracking.

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

Sincerely,

Robert E. Martin

Robert E. Martin, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 129 to DPR-44
2. Amendment No. 132 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page

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

Peach Bottom Atomic Power Station,
Units 2 and 3

cc:
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Pennsylvania Department of
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Mr. Albert R. Steel, Chairman
Board of Supervisors
Peach Bottom Township
R. D. #1
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Mr. Gary Mock
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Mr. Tom Magette
Power Plant Research Program
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Annapolis, Maryland 21401

Mr. Roland Fletcher
Department of Environment
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Baltimore, Maryland 21201

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. 129
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 12, 1987 as supplemented on October 20, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I.

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

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

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

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

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

8803210274 880303
PDR ADOCK 05000277
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 129 , 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

/s/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1988

PDI-2/LA MO'Brien 2/24/87	PDI-2/PM RMartin:mr 01/05/88	OGC SETurk 2/25/88	PDI-2/D WButler 3/3/87

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 129 , are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 129

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines. Asterisk page is provided for document completeness.

Remove	Insert
37	37
38	38
47	47*
48	48
61	61
62	62
63	63
90	90

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
1	1	Mode Switch In Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
2	1	Manual Scram		X	X	X	2 Instrument Channels	A
3	3	IRM High Flux	≤120/125 of Full Scale	X	X	(5)	8 Instrument Channels	A
4	3	IRM Inoperative		X	X	(5)	8 Instrument Channels	A
5	2	APRM High Flux	$(0.58W+62-0.58\Delta W)$ W			X	6 Instrument Channels	A or B
6	2	APRM Inoperative	(11)	X	X	X	6 Instrument Channels	A or B
7	2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
8	2	APRM High Flux in Startup	≤15% Power	X	X		6 Instrument Channels	A
9	2	High Reactor Pressure	≤1055 psig	X(9)	X	X	4 Instrument Channels	A
10	2	High Drywell Pressure	≤2 psig	X(8)	X(8)	X	4 Instrument Channels	A
11	2	Reactor Low Water Level	≥0 in. Indicated Level	X	X	X	4 Instrument Channels	A

Table 3.1.1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
12	2	High Water Level in Scram Discharge Instrument Volume	≤50 Gallons	X(2)	X	X	4 Instrument Channels	A
13	2	Turbine Condenser Low Vacuum	≥23 in. Hg. Vacuum			X	4 Instrument Channels	A or C
14	2	Main Steam Line High Radiation	≤15 X Normal Full Power Background	X	X	X	4 Instrument Channels	A
15	4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure			X(6)	8 Instrument Channels	A
16	2	Turbine Control Valve Fast Closure	500<P<850 psig Control Oil Pressure Between Fast Closure Solenoid and Disc Dump Valve			X(4)	4 Instrument Channels	A or D
17	4	Turbine Stop Valve Closure	≤10% Valve Closure			X(4)	8 Instrument Channels	A or D

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertant criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference subsection 7.2 FSAR). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

The APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other subchannel. APRM's B, D and F are arranged similarly in

3.0 BASES (Cont'd)

the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, generator load rejection, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

Instrumentation sensing drywell pressure is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds fifteen times normal background. The purpose of this scram is to limit fission product release so that 10 CFR Part 100 guidelines are not exceeded. Discharge of excessive amounts of radioactivity to the site environs is prevented by the off-gas treatment system, which provides sufficient delay time to reduce fission product release rates to well below 10 CFR 20 guidelines.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (High flux in Start-up or Refuel) system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges.

The IRM system provides protection against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
1	2 (6)	Reactor Low Water Level	≥ 0" Indicated Level (3)	4 Inst. Channels	A
2	1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
3	2	Reactor Low-Low-Low Water Level	at or above -160" indicated level (4)	4 Inst. Channels	A
4	2 (6)	High Drywell Pressure	≤ 2 psig	4 Inst. Channels	A
5	2	High Radiation Main Steam Line Tunnel	≤15 X Normal Rated Full Power Background (8)	4 Inst. Channels	B
6	2	Low Pressure Main Steam Line	≥ 850 psig (7)	4 Inst. Channels	B
7	2 (6)	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
8	2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤ 200 deg. F (9)	4 Inst. Channels	B

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
9	2	Main Steam Line Leak Detection High Temperature	≤ 200 Deg. F	4 Inst. Channels	B
10	1	Reactor Cleanup System High Flow	≤ 300% of Rated Flow	2 Inst. Channels	C
11	1	Reactor Cleanup System High Temperature	≤ 200 Deg. F.	1 Inst. Channels	E
12	2	Reactor Pressure (Feedwater Flush System Interlock)	≤ 600 psig	4 Inst. Channels	F

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken:

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate Shutdown Cooling.

   E. Isolate Reactor Water Cleanup Filter Demineralizers unless the following provision is satisifed. The RWCU Filter Demineralizer may be used (the isolation overridden) to route the reactor water to the main condenser or waste surge tank, with the high temperature trip inoperable for up to 48 hours, provided the water inlet temperature is monitored once per hour and confirmed to be below 180 degrees F.

   F. Isolate Feedwater Flush System

3. Instrument setpoint corresponds to 538 inches above vessel zero.

4. Instrument setpoint corresponds to 378 inches above vessel zero.

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in Run Mode (interlocked with Mode Switch).

8. An alarm will be tripped in the control room to alert the control room operators to an increase in the main steam line tunnel radiation level.

3.2 BASES (Cont'd)

the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. See Spec. 3.7 for Isolation Valve Closure Group. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Groups 4 and 5.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures peak at approximately 1000 degrees F and release of radioactivity to the environs is below CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel exhaust duct and along the steam line in the turbine building to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Spec. 3.7 for Valve Group. The setting is 200 degrees F for the main steam line tunnel detector. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 15 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR.

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. 132
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 12, 1987 as supplemented on October 20, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I.

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 132 , 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

/s/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1988

PDI-2/LA MO'Brien 2/24/88	PDI-2/PM RMartin:mr 01/05/88	OGC SETurk 2/25/88	PDI-2/D WButler 3/3/87

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 132, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 132

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines. Asterisk page is provided for document completeness.

Remove	Insert
37	37
38	38
40	40
47	47*
48	48
61	61
62	62
63	63
90	90

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
1	1	Mode Switch In Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
2	1	Manual Scram		X	X	X	2 Instrument Channels	A
3	3	IRM High Flux	≤120/125 of Full Scale	X	X	(5)	8 Instrument Channels	A
4	3	IRM Inoperative		X	X	(5)	8 Instrument Channels	A
5	2	APRM High Flux	(.66W+54-0.66 ΔW) FRP/MFLPD (12) (13)			X	6 Instrument Channels	A or B
6	2	APRM Inoperative	(11)	X	X	X	6 Instrument Channels	A or B
7	2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
8	2	APRM High Flux in Startup	≤15% Power	X	X		6 Instrument Channels	A
9	2	High Reactor Pressure	≤1055 psig	X(9)	X	X	4 Instrument Channels	A
10	2	High Drywell Pressure	≤2 psig	X(8)	X(8)	X	4 Instrument Channels	A
11	2	Reactor Low Water Level	≥0 in. Indicated Level	X	X	X	4 Instrument Channels	A

Table 3.1.1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
12	2	High Water Level in Scram Discharge Instrument Volume	≤50 Gallons	X(2)	X	X	4 Instrument Channels	A
13	2	Turbine Condenser Low Vacuum	≥23 in. Hg. Vacuum			X	4 Instrument Channels	A or C
14	2	Main Steam Line High Radiation	≤15 X Normal Full Power Background	X	X	X	4 Instrument Channels	A
15	4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure			X(8)	8 Instrument Channels	A
16	2	Turbine Control Valve Fast Closure	500<P<850 psig Control Oil Pressure Between Fast Closure Solenoid and Disc Dump Valve			X(4)	4 Instrument Channels	A or D
17	4	Turbine Stop Valve Closure	≤10% Valve Closure			X(4)	8 Instrument Channels	A or D

NOTES FOR TABLE 3.1.1 (Cont'd)

10. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

11. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 14 LPRM inputs of the normal complement.

12. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), where:

FRP = fraction of rated thermal power (3293 MWt).
MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for all 8 x 8 fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

W = Loop Recirculation flow in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

Delta W = the difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.66 delta W) is accomplished by correcting the flow input of the flow biased High Flux trip setting to preserve the original (two loop) relationship between APRM High Flux setpoint and recirculation drive flow or by adjusting the APRM Flux trip setting. Delta W equals zero for two loop operation.

Trip level setting is in percent of rated power (3293 MWt).

13. See Section 2.1.A.1.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertant criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference subsection 7.2 FSAR). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e, an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

The APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other subchannel. APRM's B, D and F are arranged similarly in

3.0 BASES (Cont'd)

the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, generator load rejection, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

Instrumentation sensing drywell pressure is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds fifteen times normal background. The purpose of this scram is to limit fission product release so that 10 CFR Part 100 guidelines are not exceeded. Discharge of excessive amounts of radioactivity to the site environs is prevented by the off-gas treatment system, which provides sufficient delay time to reduce fission product release rates to well below 10 CFR 20 guidelines.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (High flux in Start-up or Refuel) system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges.

The IRM system provides protection against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
1	2 (6)	Reactor Low Water Level	≥ 0" Indicated Level (3)	4 Inst. Channels	A
2	1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
3	2	Reactor Low-Low-Low Water Level	at or above -160" indicated level (4)	4 Inst. Channels	A
4	2 (6)	High Drywell Pressure	≤ 2 psig	4 Inst. Channels	A
5	2	High Radiation Main Steam Line Tunnel	≤15 X Normal Rated Full Power Background (8)	4 Inst. Channels	B
6	2	Low Pressure Main Steam Line	≥ 850 psig (7)	4 Inst. Channels	B
7	2 (5)	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
8	2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤ 200 deg. F (9)	4 Inst. Channels	B

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
9	2	Main Steam Line Leak Detection High Temperature	≤ 200 Deg. F	4 Inst. Channels	B
10	1	Reactor Cleanup System High Flow	≤ 300% of Rated Flow	2 Inst. Channels	C
11	1	Reactor Cleanup System High Temperature	≤ 200 Deg. F.	1 Inst. Channels	E
12	2	Reactor Pressure (Feedwater Flush System Interlock)	≤ 600 psig	4 Inst. Channels	F

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken:

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate Shutdown Cooling.

   E. Isolate Reactor Water Cleanup Filter Demineralizers unless the following provision is satisifed. The RWCU Filter Demineralizer may be used (the isolation overridden) to route the reactor water to the main condenser or waste surge tank, with the high temperature trip inoperable for up to 48 hours, provided the water inlet temperature is monitored once per hour and confirmed to be below 180 degrees F.

   F. Isolate Feedwater Flush System

3. Instrument setpoint corresponds to 538 inches above vessel zero.

4. Instrument setpoint corresponds to 378 inches above vessel zero.

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in Run Mode (interlocked with Mode Switch).

8. An alarm will be tripped in the control room to alert the control room operators to an increase in the main steam line tunnel radiation level.

3.2 BASES (Cont'd)

the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. See Spec. 3.7 for Isolation Valve Closure Group. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Groups 4 and 5.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures peak at approximately 1000 degrees F and release of radioactivity to the environs is below CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel exhaust duct and along the steam line in the turbine building to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Spec. 3.7 for Valve Group. The setting is 200 degrees F for the main steam line tunnel detector. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 15 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENT NOS. 129 AND 132 TO FACILITY OPERATING

LICENSE NOS. DPR-44 and DPR-56

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

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

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By letter dated February 12, 1987 as supplemented on October 20, 1987, Philadelphia Electric Company requested an amendment to Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station, Unit Nos. 2 and 3. The October 20, 1987 letter transmitted additional requested information and did not change the original application. These amendments would revise the Technical Specifications with changes related to implementation of a Hydrogen Water Chemistry (HWC) program to improve reactor water chemistry and thus to reduce the potential for intergranular stress corrosion cracking. The requested change involves changing the setpoints of the Main Steam Line Radiation Monitors (MSLRMs) from three times normal full power background (NFPB) to 15 times NFPB to support the implementation of hydrogen water chemistry. The MSLRM setpoint change is necessary since the injection of hydrogen into the feedwater radiolytically increases the N-16 volatility, thus, increasing the N-16 activity in the main steam line.

By letters dated May 27 and August 24, 1987, the licensee submitted the safety evaluation for the storage of liquid hydrogen and oxygen and additional information relating to the implementation of hydrogen water chemistry for review and approval. By separate correspondence dated September 22, 1987 the staff provided its evaluation which found the permanent hydrogen water chemistry installation to be acceptable.

2.0 EVALUATION

The main steam line radiation monitors (MSLRMs) provide reactor scram and reactor vessel and primary containment isolation signals when high-activity levels are detected in the main steam lines and serve to limit radioactivity release in the event of fuel failures. Technical Specification (TS) changes are needed to accommodate the expected increase in main steam activity levels (from increased N-16 levels in the steam phase) as a result of hydrogen injection into the primary system.

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The licensee has requested Technical Specification changes involving increasing the MSLRM setpoint from three times normal full power background (NFPB) to fifteen times NFPB to support the hydrogen water chemistry program. The Technical Specification changes for Peach Bottom Atomic Power Station Units 2 and 3 are attached and are as follows:

- Technical Specification (TS) pages 38, 48, 61, and 90 indicate the proposed change in trip setpoint of the MSLRMs from three times NFPB to 15 times NFPB.

- Note 8 of Table 3.2.A, TS page 63 is changed to delete the MSLRM alarm setpoint of 1.5 times NFPB. This permits the alarm setpoint to be adjusted based on operating experience. At the present setpoint of 1.5 times NFPB, the alarm would be on continuously during hydrogen water chemistry operation.

- On page 48 of the Technical Specifications, a change has been proposed to correct a discrepancy concerning isolation of the main condenser off-gas line by the air ejector off-gas monitor trip feature. The air ejection off-gas monitors isolation trip features were deleted by Amendment Nos. 102 and 104 for Units 2 and 3 respectively. The proposed change establishes consistency in the Technical Specifications.

- For Unit 3, Technical Specification pages 38, 40 and 61 involve proposed change for the deletion of an obsolete note that was used during the now completed Unit 3 hydrogen injection test.

- Other Technical Specification proposed changes on pages 37, 38, 61 and 62 are administrative in nature and involve the addition of item numbers to Tables for ease of referencing.

In the event of a Control Rod Drop Accident (CRDA), the MSLRMs detect high radiation levels in the main steam lines and provide signals for reactor scram and Main Steam Isolation Valve (MSIV) closure to reduce the release of fission products to the environment. The proposed MSLRM trip setpoint change from three times NFPB to 15 times NFPB will not affect the radiological consequences of a CRDA. For the proposed setpoint of 15 times NFPB the calculated dose rate at the MSLRM is about 1/4 of that resulting from a CRDA. Therefore, at the proposed setpoint, the radiation levels caused by a CRDA will still isolate the MSIVs and scram the reactor. Although the MSLRMs isolate the MSIVs and scram the reactor, the time required to reach the proposed MSLRM trip setpoint will be increased. However, this delay time in reaching the proposed MSLRM setpoint of 15 times NFPB remains within the 0.5 seconds assumed for the instrument loop response time in the UFSAR as also discussed in further detail in the licensee's letter of October 20, 1987.

In the event of an incident causing minor fuel damage such that radiation levels will not exceed the proposed MSLRM setpoint of 15 times NFPB, the downstream steam jet air ejector radiation detectors would be alarmed. These detectors have a greater sensitivity than the MSLRMs for noble gases because of a two-minute holdup period that allows for decay of N-16. The Peach Bottom Technical Specifications have established gaseous radioactivity release limits. Thus, the proposed MSLRM setpoint change will not result in offsite doses in excess of established release limits.

Therefore, the proposed Technical Specification changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of the amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 42368) on November 4, 1987 and consulted with the Commonwealth of Pennsylvania. No public comments were received. The Commonwealth of Pennsylvania had oral comments regarding the licensee's basis for selecting the revised MSLRM setpoint and on whether the alarm addressed on TS page 63 was being retained. The Commonwealth of Pennsylvania representative discussed these concerns with the licensee and the NRC staff and, on being advised of the earlier tests conducted to determine the setpoint (which were the subject of Amendment No. 106 to the Unit 2 license) and that the alarm is being retained, indicated no objection to the proposed amendments.

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

Principal Contributor: F. Witt

Dated: March 3, 1988