LICENSE AUTHORITY FILE COPY October 2, 1985 DO NOT REMOVE Posted Amdt. 115 to DPR-56

Dockets Nos. 50-277
and 50-278

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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: TECHNICAL SPECIFICATION AMENDMENTS PERTAINING TO YOUR APPLICATION DATED APRIL 19, 1984

Re: Peach Bottom Atomic Power Station, Units 2 and 3

The Commission has issued the enclosed Amendments Nos. 111 and 115 , 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 response to your application of April 19, 1984, as supplemented October 2, 1984.

The changes to the TSs (1) correct errors and establish consistency in the reactor water level setpoint values, (2) lower the main steam line isolation valve low water isolation setpoint, and (3) revise the audit frequency of the Facility Emergency Plan and implementing procedures to conform with the Commission's regulations.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Biweekly Federal Register notice.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 111 to DPR-44
2. Amendment No. 115 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
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Mr. E. G. Bauer, Jr.
Philadelphia Electric Company

Peach Bottom Atomic Power Station,
Units 2 and 3

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

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 111
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 April 19, 1984, as supplemented October 2, 1984, 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. 111, 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

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 2, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 111

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
11a	11a
12	12
15	15
21	21
61	61
63	63
72	72
79	79
80	80
89	89
90	90*
182	182
199	199
252	252

* Overleaf page included for document completeness.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
B. Core Thermal Power Limit Reactor Pressure $\leq$ 800 psia)	B. APRM Rod Block Trip Setting

SRB $\leq$ (0.66 W + 42% - 0.66 ΔW) $\frac{\text{(FRP)}}{\text{MFLPD}}$

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.

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than minus 160 inches indicated level (378 inches above vessel zero).

C. Scram and isolation--$\geq$ 538 in. above vessel zero (0" on level instruments) reactor low water level

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1 (Cont'd)

D.	Scram-- turbine stop valve closure	$\leq$10 percent
E.	Scram-- turbine control fast closure on loss of control oil pressure. $500 < P < 850$ psig.	
F.	Scram--low condenser vacuum	$\geq$ 23 inches Hg vacuum
G.	Scram--main steam line isolation	$\leq$10% valve closure
H.	Main steam isolation valve closure--nuclear system low pressure	$\geq$850 psig
I.	Core Spray & LPCI actuation--reactor low-low-low water level	$\geq$ minus 160 in. Indicated level ($\geq$ 378 inches above vessel zero)
J.	HPCI & RCIC actuation--reactor low-low water level	$\geq$ minus 48 in. Indicated level ($\geq$ 490 inches above vessel zero)
K.	Main steam isolation valve closure--reactor low-low-low water level	$\geq$ minus 160 in. Indicated level ($\geq$ 378 inches above vessel zero)

1.1.C BASES (Cont'd.)

However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit, provided scram signals are operable, is supported by the extensive plant safety analysis.

The computer provided with Peach Bottom Unit 2 has a sequence annuciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied upon to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at minus 160 inches indicated level (378 inches above vessel zero) provides adequate margin to assure sufficient cooling during shutdown conditions. This level will be continuously monitored.

E. References

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, January 1977 (NEDO-10958-A).

2. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June 1974 (NEDO-20340).

3. "General Electric Boiling Water Reactor Generic Reload Fuel Application", NEDE-24011-P-A.

2.1 BASES (Cont'd.)

C. Reactor Water Low Level Scram and Isolation (Except Main Steamlines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than the fuel cladding integrity safety limit in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 23 inches below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of less than or equal to 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

E. Turbine Control Valve Scram

The turbine control valve fast closure scram anticipates the pressure, neutron flux and heat flux increase that could result from fast closure of the turbine control valves due to a load rejection exceeding the capacity of the bypass valves or a failure in the hydraulic control system which results in a loss of oil pressure. This scram is initiated from pressure switches in the hydraulic control system which sense loss of oil pressure due to the opening of the fast acting solenoid valves or a failure in the hydraulic control system piping. Two turbine first stage pressure switches for each trip system initiate automatic bypass of the turbine control valve fast closure scram when the first stage pressure is below that required to produce 30% of rated power. Control valve closure time is approximately twice as long as that for stop valve closure.

Amendment No. 82, 111

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

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

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 satisfied. 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.

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. At a radiation level of 1.5 times the normal rated power background, 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.

9. In the event of a loss of ventilation in the main steam line tunnel area, the main steam line tunnel exhaust duct high temperature setpoint may be raised up to 250 degrees F for a period not to exceed 30 minutes to permit restoration of the ventilation flow. During the 30-minute period, an operator shall observe control room indications of the duct temperature so in the event of rapid increases (indicative of a steam line break) the operator shall promptly close the main steam line isolation valves.

Notes for Table 3.2.B

1. Whenever any CSCS subsystem is required by Section 3.5 to be operable, there shall be two operable trip systems. If the first column cannot be met for one of the trip systems, that trip system shall be placed in the tripped condition or the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Close isolation valves in RCIC subsystem.

3. Close isolation valves in HPCI subsystem.

4. Instrument set point corresponds to 378 inches above vessel zero.

5. HPCI has only one trip system for these sensors.

TABLE 3.2.G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
1	Reactor High Pressure	≤ 1120 psig	4	(2)
1	Reactor Low-Low Water Level	> -48 in. indicated level	4	(2)

Notes for Table 3.2.G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for each operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.

Amendment No. 30, 111

TABLE 4.2.A

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

	Instrument Channel (5)	Instrument Functional Test	Calibration Frequency	Instrument Check
1)	Reactor High Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2)	Reactor Low-Low-Low Water Level (7)	(1) (3)	Once/operating cycle	Once/day
3)	Main Steam High Temp.	(1) (3)	Once/operating cycle	Once/day
4)	Main Steam High Flow (7)	(1) (3)	Once/operating cycle	Once/day
5)	Main Steam Low Pressure	(1)	Once/3 months	None
6)	Reactor Water Cleanup High Flow	(1)	Once/3 months	Once/day
7)	Reactor Water Cleanup High Temp.	(1)	Once/3 months	None

	Logic System Functional Test (4) (6)	Frequency
1)	Main Steam Line Isolation Vvs. Main Steam Line Drain Vvs. Reactor Water Sample Vvs.	Once/6 months
2)	RHR - Isolation Vv. Control Shutdown Cooling Vvs. Head Spray	Once/6 months
3)	Reactor Water Cleanup Isolation	Once/6 months
4)	Drywell Isolation Vvs. TIP Withdrawal Atmospheric Control Vvs. Sump Drain Valves	Once/6 months
5)	Standby Gas Treatment System Reactor Building Isolation	Once/6 months

Amendment No. 36, 111

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out-of-service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at zero inches indicated level (538 inches above vessel zero) closes all isolation valves except those in Groups 1, 4 and 5. Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level, this trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low-low reactor water level instrumentation is set to trip when reactor water level is minus 48 inches indicated level (490 inches above vessel zero). This trip initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the reactor water level is minus 160 inches indicated level (378 inches above vessel zero). This trip closes Main Steam Line Isolation Valves, Main Steam Drain Valves and Recirc Sample Valves (Group 1), activates the remainder of the CSCS subsystem, and starts

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^{o}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^{o}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 3 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.

T.S. Change #2

NOTES FOR TABLE NO. 3.7.1

Key: O = Open
C = Closed
SC = Stays Closed
GC = Goes Closed

Note: Isolation groupings are as follows:

GROUP 1: The valves in Group 1 are actuated by any one of the following conditions:

1. Reactor vessel low-low-low water level.

2. Main steam line high radiation.

3. Main steam line high flow.

4. Main steam line space high temperature.

5. Main steam line low pressure (RUN mode only).

GROUP 2A: The valves in Group 2A are actuated by any one of the following conditions:

1. Reactor vessel low water level.

2. Reactor water cleanup system heat exchanger discharge high temperature.

3. Reactor water cleanup system suction line break.

4. Standby liquid control system actuation.

GROUP 2B: The valves in Group 2B are actuated by any one of the following conditions:

1. Reactor vessel low water level.

2. High drywell pressure.

3. Reactor high pressure of shutdown mode.

GROUP 2C: The valves in Group 2C are actuated by any one of the following conditions:

1. Reactor low water level.

2. High reactor vessel pressure, (600 PSIG)

3. High drywell pressure.

GROUP 2D: The valves in Group 2D are actuated by the following conditions:

1. High drywell pressure.
2. Reactor low water level.

GROUP 3: The valves in Group 3 are actuated by any one of the following conditions:

3.7.D & 4.7.D BASES

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

Group 1: Actuation for valves associated with the isolation of the main steam system. The main steam lines are isolated by reactor vessel low-low-low water level in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature.

Group 2: Actuation for valves associated with the isolation of the reactor auxiliary systems. Some of the reactor auxiliary systems such as the RWCU and RHR shutdown cooling systems connect into the reactor coolant boundary while others such as the drywell equipment and floor drain discharge valves do not penetrate the reactor coolant boundary. Group 2 actuation is subdivided as follows:

Group 2A - process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature at the cleanup system heat exchanger/outlet or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low level isolation is provided. An alarm of high temperature in the cleanup system area will provide an indication of suction line break resulting in manual isolation of the system. During actuation of the standby liquid control system, the cleanup system is isolated.

Group 2B - isolation valves are not normally in use and are closed by reactor vessel low water level, high drywell pressure or high reactor pressure of the shutdown mode.

Group 2C - isolation valves can only be opened when the reactor is at low pressure and the core standby cooling systems are not required. Also, since the reactor vessel could potentially be drained through these process lines, these valves are closed by low water level.

6.5.2.8 Continued

e. The Facility Emergency Plan and implementing procedures at least once per year.

f. The Facility Security Plan and implementing procedures at least once per two years.

g. The Offsite Dose Calculation Manual and implementing procedures at least once per two years.

h. The performance of activities required by the Quality Assurance Program regarding the radiological monitoring program to meet the provisions of Regulatory Guide 4.1, Revision 1, April 1975, at least once per calendar year.

i. Any other area of facility operation considered appropriate by the OSR Committee or the Vice President, Electric Production.

Authority

6.5.2.9 The OSR Committee shall report to and advise the Vice President, Electric Production on those areas of responsibility specified to Section 6.5.2.7 and 6.5.2.8.

Records

6.5.2.10 Records of OSR Committee activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each OSR Committee meeting shall be prepared, approved and forwarded to the Vice President, Electric Production within 14 days following each meeting.

b. Reports of review encompassed by Section 6.5.2.7.e,f,g, and h above, shall be prepared, approved and forwarded to the Vice President, Electric Production within 14 days following completion of the review.

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. 115
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 April 19, 1984, as supplemented October 2, 1984, 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-56 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 115, 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

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 2, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 115

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
11a	11a
12	12
15	15
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63	63
72	72
79	79
80	80
89	89
90	90*
182	182
199	199
252	252

* Overleaf page included for document completeness.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

B. Core Thermal Power Limit Reactor Pressure $\leq$ 800 psia)

B. APRM Rod Block Trip Setting

$$SRB \leq (0.66\ W + 42\% - 0.66\ \Delta W)\ \frac{(FRP)}{MFLPD}$$

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.

C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than minus 160 inches indicated level (378 inches above vessel zero).

C. Scram and isolation--> reactor low water level $\geq$ 538 in. above vessel zero (0" on level instruments)

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING		
	2.1	(Cont'd)	
	D.	Scram-- turbine stop valve closure	≤10 percent
	E.	Scram-- turbine control fast closure on loss of control oil pressure. 500<P<850 psig.	
	F.	Scram--low condenser vacuum	≥ 23 inches Hg vacuum
	G.	Scram--main steam line isolation	≤10% valve closure
	H.	Main steam isolation valve closure--nuclear system low pressure	≥850 psig
	I.	Core Spray & LPCI actuation--reactor low-low-low water level	≥ minus 160 in. Indicated level (≥ 378 inches above vessel zero)
	J.	HPCI & RCIC actuation--reactor low-low water level	≥ minus 48 in. Indicated level (≥ 490 inches above vessel zero)
	K.	Main steam isolation valve closure--reactor low-low-low water level	≥ minus 160 in. Indicated level (≥ 378 inches above vessel zero)

1.1.C BASES (Cont'd.)

However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit, provided scram signals are operable, is supported by the extensive plant safety analysis.

The computer provided with Peach Bottom Unit 3 has a sequence annuciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied upon to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at minus 160 inches indicated level (378 inches above vessel zero) provides adequate margin to assure sufficient cooling during shutdown conditions. This level will be continuously monitored.

E. References

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, January 1977 (NEDO-10958-A).

2. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June 1974 (NEDO-20340).

3. "General Electric Boiling Water Reactor Generic Reload Fuel Application", NEDE-24011-P-A.

2.1 BASES (Cont'd.)

C. Reactor Water Low Level Scram and Isolation (Except Main Steamlines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than the fuel cladding integrity safety limit in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 23 inches below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of less than or equal to 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

E. Turbine Control Valve Scram

The turbine control valve fast closure scram anticipates the pressure, neutron flux and heat flux increase that could result from fast closure of the turbine control valves due to a load rejection exceeding the capacity of the bypass valves or a failure in the hydraulic control system which results in a loss of oil pressure. This scram is initiated from pressure switches in the hydraulic control system which sense loss of oil pressure due to the opening of the fast acting solenoid valves or a failure in the hydraulic control system piping. Two turbine first stage pressure switches for each trip system initiate automatic bypass of the turbine control valve fast closure scram when the first stage pressure is below that required to produce 30% of rated power. Control valve closure time is approximately twice as long as that for stop valve closure.

Amendments Nos. ~~87~~, ~~106~~, 115

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
2 (6)	Reactor Low Water Level	$\geq$ 0" Indicated Level (3)	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	$\leq$ 75 psig	2 Inst. Channels	D
2	Reactor Low-Low-Low Water Level	at or above -160" indicated level (4)	4 Inst. Channels	A
2 (6)	High Drywell Pressure	$\leq$ 2 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	$\leq$ 3 X Normal Rated (8) (10) Full Power Background	4 Inst. Channels	B
2	Low Pressure Main Steam Line	$\geq$ 850 psig (7)	4 Inst. Channels	B
2 (5)	High Flow Main Steam Line	$\leq$ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Tunnel Exhaust Duct High Temperature	$\leq$ 200 deg. F (9)	4 Inst. Channels	B

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 satisfied. 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.

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. At a radiation level of 1.5 times the normal rated power background, 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.

9. In the event of a loss of ventilation in the main steam line tunnel area, the main steam line tunnel exhaust duct high temperature setpoint may be raised up to 250 degrees F for a period not to exceed 30 minutes to permit restoration of the ventilation flow. During the 30-minute period, an operator shall observe control room indications of the duct temperature so in the event of rapid increases (indicative of a steam line break) the operator shall promptly close the main steam line isolation valves.

Notes for Table 3.2.B

1. Whenever any CSCS subsystem is required by Section 3.5 to be operable, there shall be two operable trip systems. If the first column cannot be met for one of the trip systems, that trip system shall be placed in the tripped condition or the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Close isolation valves in RCIC subsystem.

3. Close isolation valves in HPCI subsystem.

4. Instrument set point corresponds to 378 inches above vessel zero.

5. HPCI has only one trip system for these sensors.

TABLE 3.2.G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
1	Reactor High Pressure	≤ 1120 psig	4	(2)
1	Reactor Low-Low Water Level	≥ -48 in. indicated level	4	(2)

Notes for Table 3.2.G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for each operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.

Amendment No. 29, 115

TABLE 4.2.A

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

	Instrument Channel (5)	Instrument Functional Test	Calibration Frequency	Instrument Check
1)	Reactor High Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2)	Reactor Low-Low-Low Water Level (7)	(1)(3)	Once/operating cycle	Once/day
3)	Main Steam High Temp.	(1)(3)	Once/operating cycle	Once/day
4)	Main Steam High Flow (7)	(1)(3)	Once/operating cycle	Once/day
5)	Main Steam Low Pressure	(1)	Once/3 months	None
6)	Reactor Water Cleanup High Flow	(1)	Once/3 months	Once/day
7)	Reactor Water Cleanup High Temp.	(1)	Once/3 months	None

	Logic System Functional Test (4) (6)	Frequency
1)	Main Steam Line Isolation Vvs. Main Steam Line Drain Vvs. Reactor Water Sample Vvs.	Once/6 months
2)	RHR - Isolation Vv. Control Shutdown Cooling Vvs. Head Spray	Once/6 months
3)	Reactor Water Cleanup Isolation	Once/6 months
4)	Drywell Isolation Vvs. TIP Withdrawal Atmospheric Control Vvs. Sump Drain Valves	Once/6 months
5)	Standby Gas Treatment System Reactor Building Isolation	Once/6 months

Amendment No. ~~29~~, 115

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out-of-service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at zero inches indicated level (538 inches above vessel zero) closes all isolation valves except those in Groups 1, 4 and 5. Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level, this trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low-low reactor water level instrumentation is set to trip when reactor water level is minus 48 inches indicated level (490 inches above vessel zero). This trip initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the reactor water level is minus 160 inches indicated level (378 inches above vessel zero). This trip closes Main Steam Line Isolation Valves, Main Steam Drain Valves and Recirc Sample Valves (Group 1), activates the remainder of the CSCS subsystem, and starts

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^{o}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^{o}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 3 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.

T.S. Change #2

NOTES FOR TABLE NO. 3.7.1

Key: O = Open
C = Closed
SC = Stays Closed
GC = Goes Closed

Note: Isolation groupings are as follows:

GROUP 1: The valves in Group 1 are actuated by any one of the following conditions:

1. Reactor vessel low-low-low water level.
2. Main steam line high radiation.
3. Main steam line high flow.
4. Main steam line space high temperature.
5. Main steam line low pressure (RUN mode only).

GROUP 2A: The valves in Group 2A are actuated by any one of the following conditions:

1. Reactor vessel low water level.
2. Reactor water cleanup system heat exchanger discharge high temperature.
3. Reactor water cleanup system suction line break.
4. Standby liquid control system actuation.

GROUP 2B: The valves in Group 2B are actuated by any one of the following conditions:

1. Reactor vessel low water level.
2. High drywell pressure.
3. Reactor high pressure of shutdown mode.

GROUP 2C: The valves in Group 2C are actuated by any one of the following conditions:

1. Reactor low water level.
2. High reactor vessel pressure, (600 PSIG)
3. High drywell pressure.

GROUP 2D: The valves in Group 2D are actuated by the following conditions:

1. High drywell pressure.
2. Reactor low water level.

OUP 3: The valves in Group 3 are actuated by any one of the following conditions:

3.7.D & 4.7.D BASES

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

Group 1: Actuation for valves associated with the isolation of the main steam system. The main steam lines are isolated by reactor vessel low-low-low water level in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature.

Group 2: Actuation for valves associated with the isolation of the reactor auxiliary systems. Some of the reactor auxiliary systems such as the RWCU and RHR shutdown cooling systems connect into the reactor coolant boundary while others such as the drywell equipment and floor drain discharge valves do not penetrate the reactor coolant boundary. Group 2 actuation is subdivided as follows:

Group 2A - process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature at the cleanup system heat exchanger/outlet or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low level isolation is provided. An alarm of high temperature in the cleanup system area will provide an indication of suction line break resulting in manual isolation of the system. During actuation of the standby liquid control system, the cleanup system is isolated.

Group 2B - isolation valves are not normally in use and are closed by reactor vessel low water level, high drywell pressure or high reactor pressure of the shutdown mode.

Group 2C - isolation valves can only be opened when the reactor is at low pressure and the core standby cooling systems are not required. Also, since the reactor vessel could potentially be drained through these process lines, these valves are closed by low water level.

6.5.2.8 Continued

e. The Facility Emergency Plan and implementing procedures at least once per year.

f. The Facility Security Plan and implementing procedures at least once per two years.

g. The Offsite Dose Calculation Manual and implementing procedures at least once per two years.

h. The performance of activities required by the Quality Assurance Program regarding the radiological monitoring program to meet the provisions of Regulatory Guide 4.1, Revision 1, April 1975, at least once per calendar year.

i. Any other area of facility operation considered appropriate by the OSR Committee or the Vice President, Electric Production.

Authority

6.5.2.9 The OSR Committee shall report to and advise the Vice President, Electric Production on those areas of responsibility specified to Section 6.5.2.7 and 6.5.2.8.

Records

6.5.2.10 Records of OSR Committee activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each OSR Committee meeting shall be prepared, approved and forwarded to the Vice President, Electric Production within 14 days following each meeting.

b. Reports of review encompassed by Section 6.5.2.7.e,f,g, and h above, shall be prepared, approved and forwarded to the Vice President, Electric Production within 14 days following completion of the review.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENTS NOS. 111 AND 115 TO FACILITY OPERATING LICENSES NOS. DPR-44 AND DPR-56

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

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

DOCKETS NOS. 50-277 AND 50-278

INTRODUCTION

In a letter dated April 19, 1984, as supplemented October 2, 1984, the licensee (Philadelphia Electric Company (PECO)) proposed changes to the Peach Bottom Atomic Power Station, Units 2 and 3 Technical Specifications (TSs). The proposed changes reflect the PECO effort to: (1) correct errors and establish consistency in the reactor water level setpoint values, (2) lower the main steam line isolation valve low water isolation setpoint from low-low to low-low-low reactor water level, and (3) revise the audit frequency of the Facility Emergency Plan and implement procedures to conform with the Commission's regulations. A discussion of each of the proposed changes is presented below:

1. Reactor Water Level Setpoint Values

a) The current TSs specify reactor water level setpoints to three different reference points, e.g., in terms of inches above top of active fuel, inches above vessel zero, and instrument indicated level. This inconsistency results in multiple values that are confusing to the reader and may result in possible misinterpretation of the specifications. The proposed changes would establish consistency by identifying the setpoint in terms of the "instrument indicated level," with reference to the "inches above vessel zero."

b) The current TS on page 12 identifies the Core Spray and LPCI actuation setpoint as minus 159.5 inches indicated level (low-low-low level) and 378 inches above vessel zero. Tables 7.4.2, 7.4.3 and 7.4.4 of the Peach Bottom Final Safety Analysis Report, as well as design information provided by General Electric Company, confirms the 378 inch value. However, since the level instrumentation reference zero is set at 538 inches above vessel zero, the indicated level should be minus 160 inches, not 159.5 inches. General Electric Company also confirms this minor error.

c) The current TSs on pages 12, 61 and 79 identify either minus 49.5 or minus 49.0 inches indicated level (low-low level) as the setpoints for HPCI/RCIC actuation, recirculation pump trip, and primary containment isolation. According to Tables 7.3.2 and 7.4.1 of the Peach Bottom Final Safety Analysis Report, as well as design information provided by General Electric Company, the correct value should be minus 48 inches indicated level which corresponds with 490 inches above vessel zero. The low-low level on page 64 of the TSs was originally specified at minus 49 inches. The origin of this error appears to be an error during the preparation of the original license. The proposed change would specify the correct value of minus 48 inches indicated level.

d) The proposed change on page 21 of the TS Bases would correctly state the relationship between the scram setting and the normal operating level. The normal operating level is at plus 23 inches indicated level, which is approximately the midpoint between the reactor high water level trip (trips High Pressure Coolant Injection, Reactor Core Isolation Cooling, and Turbine Stop Valves) set at plus 45 inches indicated level and the reactor low water level trip (scram setpoint) set at zero inches indicated level, in order to minimize spurious trips due to feedwater transients. Therefore, the scram setting is approximately 23 inches, not 31 inches, below the normal operating range.

e) The final change involving reactor water levels specified in the TSs deals with the minimum water level required in the shutdown condition. The Standard Technical Specifications for General Electric Boiling Water Reactors (NUREG-0123, Rev. 3, page 2-2) require the reactor water level to be maintained above the top of the active irradiated fuel in the shutdown mode. The Peach Bottom TSs had conservatively set this safety limit to correspond with the low-low-low level setpoint (378 inches above vessel zero). Based on the original top of active fuel elevation of 360.3 inches above vessel zero, a value of 17.7 inches above the active fuel was specified. The specification erroneously specifies 17.7 inches as a result of a typographical error. The proposed change would correct this error and identify the safety limit in terms of the format described in (a) above.

2. Lowering the Main Steam Line Isolation Valve (MSIV) Low Water Isolation Setpoint

In response to the NRC approved Mark I containment modification program, PECO made design modifications which lowered the water level trip setpoints for MSIV closure. In its Amendment Application dated April 19, 1984, PECO requested a change in its TSs to reflect the lowering of the MSIV water level trip from level two (low-low) to level one (low-low-low).

The lower MSIV water level trip causes the MSIV closure actuation to be changed from a reactor water level two signal to a reactor water level one signal. The design modification will maintain the main condenser available for a longer time following reactor trip. This allows more energy to be released to the main condenser and will result in a slower repressurization rate. The lower MSIV water level trip reduces isolations and SRV challenges.

3. Revision of the Audit Frequency for Emergency Plans and Procedures

The current TSs (Section 6.5.2.8.e) require that the Facility Emergency Plan and implementing procedures be audited at least once per two years. The NRC issued Generic Letter 82-17 ("Inconsistency between Requirements of 10 CFR 50.54(t) and Standard Technical Specifications for Performing Audits of Emergency Preparedness Programs" dated October 1, 1982) requesting that licensees revise their TSs for performing audits on Emergency Plans and implementing procedures at least once per year rather than once per two years to bring the TSs into conformance with the requirements of 10 CFR 50.54(t). The proposed TS change makes this requested change.

EVALUATION

The licensee's proposed changes as discussed above (1a through 1e) are influenced by the modification from the 7 x 7 fuel rod array installed at the time of licensing to the 8 x 8 array in use today. The licensee states that the dimensional change associated with the new fuel design does not impact the ECCS evaluations for Peach Bottom since the values for low-low and low-low-low setpoints used in the ECCS analysis are 486.5 and 366.4 inches above vessel zero, respectively. Although the changes appear administrative in nature, from a review of the justification provided in the licensee's evaluation, we were concerned that the setpoints developed for earlier reactor core fuel arrangements may not be conservative for the current fuel rod array and fuel column length, and that sufficient allowances may not have been provided in the setpoints to account for instrument bias and uncertainty.

It was not clear from a review of the licensee's submittal of April 19, 1984, whether the ECCS evaluation for both fuel arrangements uses 486.5 and 366.4 inches as inputs to the calculations and if these values that were developed for the earlier core fuel arrangement would be conservative for the new fuel rod array and column length or whether there are other reasons why the new fuel design does not impact the analysis. Accordingly, we requested (1) the licensee either confirm in the safety analysis that the values quoted for the low-low and low-low-low setpoints (486.5 and 366.4 inches above vessel zero respectively) are the analytical values used in the safety analysis for the current fuel rod array and fuel column length as the point where protective action is initiated to demonstrate the design basis accidents or transients will not exceed specified safety limits, allowing for modeling uncertainties and bias or justify their absence, and (2) the licensee confirm that sufficient allowance is provided in the setpoints to account for instrument drift and uncertainty.

In a letter dated October 2, 1984, the licensee provided additional information concerning the Amendment application that resolved these concerns.

The proposed changes to the reactor water level setpoints maintain the low-low level (level 2) and low-low-low levels (level 1) at 490 and 378 inches above vessel zero, respectively. The dimensional change associated with the new fuel design is considered in the ECCS reload evaluations for Peach Bottom 2 and 3. The values for low-low and low-low-low setpoints used in the ECCS analysis, 486.5 and 366.4 inches above vessel zero, respectively, provide sufficient margin to compensate for instrument error and calibration accuracy. Additionally, the licensee confirmed that there is sufficient allowance for setpoint drift between the nominal trip setpoint of the instrument and the TS value.

We find the above proposed changes to the Peach Bottom Atomic Power Station, Units 2 and 3, TSs pertaining to reactor water level permit the operation of the facilities in a manner that is consistent with the licensing basis and the accident analysis. Based on the above, we conclude that the proposed TS changes concerning reactor water level are acceptable.

The lowering of MSIV water level trip setpoints as discussed in Item 2 above is in response to the approved NRC design modifications under the Commission's Mark I Containment Program. These design modifications and, therefore, the proposed corresponding lowering of the trip setpoints will not adversely affect the plants' performance or safety margins. We, therefore, find the proposed change acceptable.

The proposed change in the Emergency Plan audit requirements (Item 3 above) is responsive to an NRC request (Generic Letter 82-17) and would bring the Peach Bottom TSs into conformance with the Commission's regulations [10 CFR 50.54(t)]. We, therefore, find this change to be acceptable. The licensee notes in its application that an annual audit of the Peach Bottom emergency preparedness program has already been implemented in accordance to 10 CFR 50.54(t).

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. The amendments also relate to changes in an administrative requirement. 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) and (10). 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.

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: October 2, 1985

The following NRC personnel have contributed to this Safety Evaluation: J. Mauck, K. Desai, and G. Gears