

May 21, 1990

Docket Nos. 50-277/278

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Mr. George A. Hunger, Jr.
 Director-Licensing, MC 5-2A-5
 Philadelphia Electric Company
 Nuclear Group Headquarters
 Correspondence Control Desk
 P.O. Box No. 195
 Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

SUBJECT: CYCLE-SPECIFIC PARAMETER LIMITS (TAC NOS. 76188 AND 76189)

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

The Commission has issued the enclosed Amendments Nos. 154 and 155 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 (TS) in response to your application dated March 8, 1990 as supplemented on April 26, 1990. The supplemental letter proposed editorial and administrative revisions to the TS changes proposed in the original submittal. The staff has determined that these additional changes do not affect the proposed no significant hazards determination.

These amendments involve removal of cycle-specific parameter limits from the Technical Specifications in accordance with the guidance provided in NRC Generic Letter 88-16,

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/

Gene Y. Suh, Project Manager
 Project Directorate I-2
 Division of Reactor Projects - I/II
 Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 154 to DPR-44
2. Amendment No. 155 to DPR-56
3. Safety Evaluation

cc w/enclosures:
 See next page

[76188/9 AM]

PDI-2/PA
 MO'Brien
 5/18/90

PDI-2/PM
 GSuh:mj
 5/18/90

OGC
 M. Young
 5/11/90

PDI-2/D
 WButler
 5/21/90

C/P-1
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9006120249 900521
 PDR ADDCK 05000277
 PDC



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

May 21, 1990

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Sincerely,

A handwritten signature in cursive script that reads "Gene Y. Suh".

Gene Y. Suh, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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See next page

Mr. George A. Hunger, Jr.
Philadelphia Electric Company

Peach Bottom Atomic Power Station,
Units 2 and 3

cc:

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

Single Point of Contact
P. O. Box 11880
Harrisburg, Pennsylvania 17108-1880

Philadelphia Electric Company
ATTN: Mr. D. M. Smith, Vice President
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

Mr. Thomas M. Gerusky, Director
Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Philadelphia Electric Company
ATTN: Regulatory Engineer, A1-2S
Peach Bottom Atomic Power Station
Route 1, Box 208
Delta, Pennsylvania 17314

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

Resident Inspector
U.S. Nuclear Regulatory Commission
Peach Bottom Atomic Power Station
P.O. Box 399
Delta, Pennsylvania 17314

Public Service Commission of Maryland
Engineering Division
ATTN: Chief Engineer
231 E. Baltimore Street
Baltimore, MD 21202-3486

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. Tom Magette
Power Plant Research Program
Department of Natural Resources
B-3
Tawes State Office Building
Annapolis, Maryland 21401

Mr. Roland Fletcher
Department of Environment
201 West Preston Street
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. 154
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 March 8, 1990 as supplemented on April 26, 1990, 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:

9006120251 900521
PDR ADDCK 05000277
P FDC

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 154, 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: May 21, 1990

PDI-2/IA
MD/Bren
5/18/90

PDI-2/PM
GSuh:mj
5/17/90

OGC
M. Young
5/11/90

PDI-2/D
WButler
5/21/90




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FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **May 21, 1990**

ATTACHMENT TO LICENSE AMENDMENT NO. 154

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.

<u>Remove</u>	<u>Insert</u>
iv	iv
iva	iva
vi	vi
1	1
10	10
11a	11a
40	40
73	73
74	74
74a	74a
133a	133a
133b	133b
133c	133c
133d	133d
133e	-
140	140
140a	140a
140b	140b
141a	141a
141b	141b
142	142
142a	-

ATTACHMENT TO LICENSE AMENDMENT NO. 154

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

<u>Remove</u>	<u>Insert</u>
142a-1	-
142a-2	-
142a-3	-
142a-4	-
142a-5	-
142d	142d
142g	142g
142h	-
142i	-
142j	-
142k	-
142l	-
142m	-
142n	-
256	256
-	256a

PBAPS
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PBAPS

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud with the vessel head removed and fuel in the vessel.

Normal control rod movement with the control drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a core alteration.

Average Planar Linear Heat Generation Rate (APLHGR) - The APLHGR shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod, for all the fuel rods in the specific bundle at the specific height, divided by the number of fuel rods in the fuel bundle at that height.

Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

Cold Condition - Reactor coolant temperature equal to or less than 212 F.

Cold Shutdown - The reactor is in the shutdown mode, the reactor coolant temperature equal to or less than 212 F, and the reactor vessel is vented to atmosphere.

Core Operating Limits Report (COLR) - The COLR is the unit-specific document that provides the core operating limits for the current Operating Cycle. These cycle-specific core operating limits shall be determined for each Operating Cycle in accordance with specification 6.9.1.e. Plant operation within these limits is addressed in individual Specifications.

Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958).

Dose Equivalent I-131 - That concentration of I-131 (Ci/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present.

PBAPS

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

2.1.A (Cont'd)

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows.

$$S \leq (0.58W + 62\% - 0.58\Delta W) \frac{(FRP)}{MFLPD}$$

where,

FRP = fraction of rated thermal power (3293 Mwt)

MFLPD = maximum fraction of limiting power density where the limiting power density is the value specified in the CORE OPERATING LIMITS REPORT.

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.

2. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.
3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

PBAPS

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

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

B. APRM Rod Block Trip Setting
SRB \leq (0.58 W + 50% - 0.58 Δ 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 the value
specified in the CORE
OPERATING LIMITS REPORT.

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--> 538 in.
reactor low water level above vessel
zero (0" on
level
instruments).

PBAPS

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 the value specified in the CORE OPERATING LIMITS REPORT.

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.

ΔW = 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 (-0.58 ΔW) 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. $\Delta W = 0$ for two loop operation.

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

13. See Section 2.1.A.1.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
4 (2)	APRM Upscale (Flow Biased)	$\leq \frac{(0.58W+50-0.58\Delta W) \times FRP}{MFLPD}$	6 Inst. Channels	(10)
4	APRM Upscale (Startup Mode)	$\leq 12\%$	6 Inst. Channels	(10)
4	APRM Downscale	≥ 2.5 indicated on scale	6 Inst. Channels	(10)
1 (2)(7)(11)	Rod Block Monitor (Flow Biased)	$\leq \frac{(0.66W+(N-66)-0.66\Delta W) \times FRP}{MFLPD}$ with a maximum of $\leq N\%$	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	≥ 2.5 indicated on scale	2 Inst. Channels	(1)
6	IRM Downscale (3)	≥ 2.5 indicated on scale	8 Inst. Channels	(10)
6	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(10)
6	IRM Upscale	≤ 108 indicated on scale	8 Inst. Channels	(10)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5)(6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Level	≤ 25 gallons	1 Inst. Channel	(9)

Amendment No. 23, 34, 36, 42, 48, 70, 78, 88, 91, 104, 123, 154 -73-

PBAPS

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. The equation for Trip Level Setting 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 the value specified in the CORE OPERATING LIMITS REPORT.

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.

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

ΔW is 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 is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between the rod block setpoint and recirculation drive flow, or by adjusting the rod block setting. $\Delta W = 0$ for two loop operation.

3. IRM downscale is bypassed when it is on its lowest range.
4. This function is bypassed when the count rate is ≥ 100 cps.
5. One of the four SRM inputs may be bypassed.
6. This SRM function is bypassed when the IRM range switches are on range 8 or above.
7. The trip is bypassed when the reactor power is $\leq 30\%$.
8. This function is bypassed when the mode switch is placed in Run.

PBAPS

NOTES FOR TABLE 3.2.C (Cont.)

9. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn.
10. For the Startup (for IRM rod block) and the Run (for APRM rod block) positions of the Reactor Mode Selector Switch and with the number of OPERABLE channels:
 - a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.
 - b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.
11. The value of N is specified in the CORE OPERATING LIMITS REPORT.

PBAPS

LIMITING CONDITIONS FOR OPERATION3.5.I Average Planar LHGR

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limit for the most limiting lattice (excluding natural uranium) specified in the CORE OPERATING LIMITS REPORT during two recirculation loop operations. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the reduction factors specified in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore ALPHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.J Local LHGR

During power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed design LHGR.

$$\text{LHGR} \leq \text{LHGRd}$$

$$\text{LHGRd} = \text{Design LHGR}$$

The values for Design LHGR for each fuel type are specified in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS4.5.I Average Planar LHGR

The APLGHR for each type of fuel as a function of average planar exposure shall be checked daily during reactor operation at $\geq 25\%$ rated thermal power.

4.5.J Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq 25\%$ rated thermal power.

PBAPS

LIMITING CONDITIONS FOR OPERATION3.5.J Local LHGR (Cont'd)

If at any time during operation it is determined by normal surveillance that limiting value for LHGR is being exceeded, action shall be initiated within one (1) hour to restore LHGR to within prescribed limits. If the LHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless LHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

1. During power operation the MCPR for the applicable incremental cycle core average exposure and for each type of fuel shall be equal to or greater than the value given in Specification 3.5.K.2 or 3.5.K.3 times K_f , where K_f is as specified in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within one (1) hour to restore MCPR to within prescribed limits. If the MCPR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless MCPR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at >25% rated thermal power.
2. Except as provided in Specification 3.5.K.3, the verification of the applicability of 3.5.K.2.a Operating Limit MCPR Values shall be performed every 120 operating days by scram time testing 19 or more control rods on a rotation basis and performing the following:

a. The average scram time to the 20% insertion position shall be:

$$\tau_{ave} \leq \tau_B$$

b. The average scram time to the 20% insertion position is determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^n N_i \tau_i}{\sum_{i=1}^n N_i}$$

where: n = number of surveillance tests performed to date in the cycle.

PBAPS

LIMITING CONDITIONS FOR OPERATION3.5.K Minimum Critical Power Ratio (MCPR) (Cont'd)

2. Except as specified in 3.5.K.3, the Operating Limit MCPR Values shall be as specified in the CORE OPERATING LIMITS REPORT for when

a) requirement 4.5.K.2.a is met, and for when

b) requirement 4.5.K.2.a is not met, where:

$$\tau = \frac{\tau_{ave} - \tau_B}{0.90 - \tau_B}$$

3. If the Surveillance Requirement of Section 4.5.K.2 to scram time test control rods is not performed, the Operating Limit MCPR values shall be as specified in the CORE OPERATING LIMITS REPORT for this condition.

SURVEILLANCE REQUIREMENTS4.5.K Minimum Critical Power Ratio (MCPR) (Cont'd)

N_i = number of active control rods measured in the i th surveillance test.

τ_i = average scram time to the 20% insertion position of all rods measured in the i th surveillance test.

c. The adjusted analysis mean scram time (τ_B) is calculated as follows:

$$\tau_B = \mu + 1.65 \left(\frac{N_1}{\sum_{i=1}^{N_1} \tau_i} \right)^{1/2} \sigma$$

Where:

μ = mean of the distribution for average scram insertion time to the 20% position = 0.694 sec.

N_1 = total number of active control rods measured in specification A.3.C.1.

σ = standard deviation of the distribution for average scram insertion time to the 20% position = 0.016.

PBAPS

Tables 3.5.K.2 and 3.5.K.3 have been removed from former Technical Specification pages 133d and 133e, respectively, and the associated information has been relocated to the Core Operating Limits Report.

PBAPS

3.5 BASES (Cont'd.)H. Engineered Safeguards Compartments Cooling and Ventilation

One unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicated that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured. Ventilation associated with the High Pressure Service Water Pumps is also associated with the Emergency Service Water pumps, and is specified in Specification 3.9.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR Part 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent, secondarily, on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The Technical Specification APLHGR is the LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in the applicable figure for each fuel type in the CORE OPERATING LIMITS REPORT.

Only the most limiting APLHGR operating limits are shown in the figures for the multiple lattice fuel types. Compliance with the lattice-specific APLHGR limits is ensured by using the process computer. When an alternate method to the process computer is required (i.e. hand calculations and/or alternate computer simulation), the most limiting lattice APLHGR limit for each fuel type shall be applied to every lattice of that fuel type.

The calculational procedure used to establish the APLHGR is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (G.E.) calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. A complete discussion of each code employed in the analysis is presented in Reference 4. Input and model changes in the Peach Bottom loss-of-coolant analysis which are different from the previous analyses performed with Reference 4 are described in detail in Reference 8. These changes to the analysis include: (1) consideration of the counter current flow limiting (CCFL) effect, (2) corrected code inputs, and (3) the effect of drilling alternate flow paths in the bundle lower tie plate.

PBAPS

3.5 BASES (Cont'd)J. Local LHGR

This specification assures that the linear heat generation rate in any 8X8 fuel rod is less than the design linear heat generation. The maximum LHGR shall be checked daily during reactor operation at > 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be at the design LHGR below 25% rated thermal power, the peak local LHGR must be a factor of approximately ten (10) greater than the average LHGR which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and Reference 7. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not violated during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The transients evaluated are as described in Reference 7.

PBAPS

3.5.K BASES (Cont'd)

The largest reduction in critical power ratio is then added to the fuel cladding integrity safety limit MCPR to establish the MCPR Operating Limit for each fuel type.

Analysis of the abnormal operational transients is presented in Reference 7. Input data and operating conditions used in this analysis are shown in Reference 7 and in the Supplemental Reload Licensing Analysis.

3.5.L Average Planar LHGR (APLHGR), Local LHGR and Minimum Critical Power Ratio (MCPR)

In the event that the calculated value of APLHGR, LHGR or MCPR exceeds its limiting value, a determination is made to ascertain the cause and initiate corrective actions to restore the value to within prescribed limits. The status of all indicated limiting fuel bundles is reviewed as well as input data associated with the limiting values such as power distribution, instrumentation data (Traversing In-Core Probe - TIP, Local Power Range Monitor - LPRM, and reactor heat balance instrumentation), control rod configuration, etc., in order to determine whether the calculated values are valid.

In the event that the review indicates that the calculated value exceeding limits is valid, corrective action is immediately undertaken to restore the value to within prescribed limits. Following corrective action, which may involve alterations to the control rod configuration and consequently changes to the core power distribution, revised instrumentation data, including changes to the relative neutron flux distribution, for up to 43 in-core locations is obtained and the power distribution, APLHGR, LHGR and MCPR calculated. Corrective action is initiated within one hour of an indicated value exceeding limits and verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication.

In the event that the calculated value of APLHGR, LHGR or MCPR exceeding its limiting value is not valid, i.e., due to an erroneous instrumentation indication, etc., corrective action is initiated within one hour of an indicated value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

PBAPS

4.5.K Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

4.5.L MCPR Limits for Core Flows Other Than Rated

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves in the CORE OPERATING LIMITS REPORT were developed generically and are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow rate (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of the core flow, divided by the operating limit MCPR determines the K_f .

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

PBAPS

4.5.L MCPR Limits for Core Flows Other Than Rated (Cont'd.)

The K_f factors specified in the CORE OPERATING LIMITS REPORT are acceptable for Peach Bottom operation because the operating limit MCPR is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f .

PBAPS

The following Figures have been removed from the Technical Specifications and the associated information has been relocated to the Core Operating Limits Report:

Figure 3.5.K.1-1, former page 142
Figure 3.5.K.2, former page 142a
Figure 3.5.K.1-2, former page 142a-1
Figure 3.5.K.1-3, former page 142a-2
Figure 3.5.K.2-1, former page 142a-3
Figure 3.5.K.2-2, former page 142a-4
Figure 3.5.K.2-3, former page 142a-5

PBAPS

Figure 3.5.1.E has been removed from this page of the Technical Specifications and the associated information has been relocated to the Core Operating Limits Report.

PBAPS

The following Figures have been removed from the Technical Specifications and the associated information has been relocated to the Core Operating Limits Report:

- Figure 3.5.1.H, former page 142g
- Figure 3.5.1.I, former page 142h
- Figure 3.5.1.J, former page 142i
- Figure 3.5.1.K, former page 142j
- Figure 3.5.1.L, former page 142k
- Figure 3.5.1.M, former page 142l
- Figure 3.5.1.N, former page 142m
- Figure 3.5.1.O, former page 142n

PBAPS

6.9.1 Routine Reports (Cont'd)c. Annual Safety/Relief Valve Report

Describe all challenges to the primary coolant system safety and relief valves. Challenges are defined as the automatic opening of the primary coolant safety or relief valves in response to high reactor pressure.

d. Monthly Operating Report

Routine reports of operating statistics and shutdown experience and a narrative summary of the operating experience shall be submitted on a monthly basis to the Office of Management and Program Analysis (or its successor), U.S. Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the appropriate Regional Office, to be submitted no later than the 15th of the month following the calendar month covered by the report.

e. Core Operating Limits Report

(1) Core operating limits shall be established and shall be documented in the CORE OPERATING LIMITS REPORT prior to each Operating Cycle, or prior to any remaining portion of an Operating Cycle, for the following:

- a. The APLHGR for Specification 3.5.I,
- b. The MCPR for Specification 3.5.K,
- c. The K_f core flow adjustment factor for Specification 3.5.K,
- d. The LHGR for Specification 3.5.J,
- e. The upscale flow biased Rod Block Monitor setpoint and the upscale high flow clamped Rod Block monitor setpoint of Specification 3.2.C.

(2) The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents as amended and approved:

- a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version)
- b. Philadelphia Electric Company Methodologies as described in:
 - (1) PECO-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code"

PBAPS

6.9.1 Routine Reports (Cont'd)

- (2) PECO-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECo)"
 - (3) PECO-FMS-0003-A, "Steady-State Fuel Performance Methods Report"
 - (4) PECO-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis"
 - (5) PECO-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"
- (3) The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.
 - (4) The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be submitted upon issuance for each Operating Cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.



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. 155
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 March 8, 1990 as supplemented on April 26, 1990, 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. 155, 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: **May 21, 1990**

PDI-2/A
MOL Brien
5/17/90

PDI-2/PM
GSuh:mj
5/17/90

OGC
m...
5/11/90

PDI-2/D
WButler
5/21/90

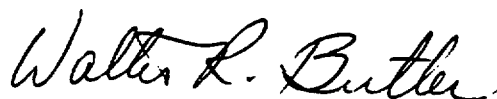
WB

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 155, 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, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **May 21, 1990**

ATTACHMENT TO LICENSE AMENDMENT NO. 155

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.

<u>Remove</u>	<u>Insert</u>
iv	iv
-	iva
vi	vi
1	1
10	10
11a	11a
40	40
73	73
74	74
74a	74a
133a	133a
133b	133b
133c	133c
133d	133d
133e	-
140	140
140a	140a
141a	141a
141b	141b
142	142
142a	-

ATTACHMENT TO LICENSE AMENDMENT NO. 155

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

<u>Remove</u>	<u>Insert</u>
142d	-
142e	-
142f	-
142g	-
142h	-
142i	-
142j	-
256	256
-	256a

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Amendment No. 14, 41, 45, 46, 62, 79,
 92, 104, 107, 114, 126, -iv-
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PBAPS

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud with the vessel head removed and fuel in the vessel.

Normal control rod movement with the control drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a core alteration.

Average Planar Linear Heat Generation Rate (APLHGR) - The APLHGR shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod, for all the fuel rods in the specific bundle at the specific height, divided by the number of fuel rods in the fuel bundle at that height.

Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

Cold Condition - Reactor coolant temperature equal to or less than 212 F.

Cold Shutdown - The reactor is in the shutdown mode, the reactor coolant temperature equal to or less than 212 F, and the reactor vessel is vented to atmosphere.

Core Operating Limits Report (COLR) - The COLR is the unit-specific document that provides the core operating limits for the current Operating Cycle. These cycle-specific core operating limits shall be determined for each Operating Cycle in accordance with specification 6.9.1.e. Plant operation within these limits is addressed in individual Specifications.

Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958).

Dose Equivalent I-131 - That concentration of I-131 (Ci/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present.

PBAPS

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

2.1.A (Cont'd)

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows.

$$S \leq (0.58W + 62\% - 0.58\Delta W) \frac{(FRP)}{MFLPD}$$

where,

FRP = fraction of rated thermal power (3293 Mwt)

MFLPD = maximum fraction of limiting power density where the limiting power density is the value specified in the CORE OPERATING LIMITS REPORT.

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.

2. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.
3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

PBAPS

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

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

B. APRM Rod Block Trip Setting
SRB \leq (0.58 W + 50% - 0.58 Δ 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 the value
specified in the CORE
OPERATING LIMITS REPORT.

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-- \geq 538 in.
reactor low water level above vessel
zero (0" on
level
instruments).

PBAPS

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 the value
 specified in the CORE OPERATING
 LIMITS REPORT.

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 (-0.58 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.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
4 (2)	APRM Upscale (Flow Biased)	$\leq (0.58W + 50 - 0.58\Delta W) \times \frac{FRP}{MFLPD}$	6 Inst. Channels	(10)
4	APRM Upscale (Startup Mode)	$\leq 12\%$	6 Inst. Channels	(10)
4	APRM Downscale	≥ 2.5 indicated on scale	6 Inst. Channels	(10)
1 (2)(7)(11)	Rod Block Monitor (Flow Biased)	$\leq (0.66W + (N-66) - 0.66\Delta W) \times \frac{FRP}{MFLPD}$ with a maximum of $\leq N\%$	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	≥ 2.5 indicated on scale	2 Inst. Channels	(1)
6	IRM Downscale (3)	≥ 2.5 indicated on scale	8 Inst. Channels	(10)
6	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(10)
6	IRM Upscale	≤ 108 indicated on scale	8 Inst. Channels	(10)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5)(6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Level	≤ 25 gallons	1 Inst. Channel	(9)

Amendment No. 33, 62, 77, 80, 93, 100, -73-
150, 155

PBAPS

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.
2. The equation for Trip Level Setting 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 the value specified in the CORE OPERATING LIMITS REPORT.

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.

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

ΔW is 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 is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between the rod block setpoint and recirculation drive flow, or by adjusting the rod block setting. $\Delta W = 0$ for two loop operation.

3. IRM downscale is bypassed when it is on its lowest range.
4. This function is bypassed when the count rate is ≥ 100 cps.
5. One of the four SRM inputs may be bypassed.
6. This SRM function is bypassed when the IRM range switches are on range 8 or above.
7. The trip is bypassed when the reactor power is $\leq 30\%$.
8. This function is bypassed when the mode switch is placed in Run.

PBAPS

NOTES FOR TABLE 3.2.C (Cont.)

9. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn.
10. For the Startup (for IRM rod block) and the Run (for APRM rod block) positions of the Reactor Mode Selector Switch and with the number of OPERABLE channels:
 - a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.
 - b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.
11. The value of N is specified in the CORE OPERATING LIMITS REPORT.

PBAPS

LIMITING CONDITIONS FOR OPERATION3.5.I Average Planar LHGR

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limit for the most limiting lattice (excluding natural uranium) specified in the CORE OPERATING LIMITS REPORT during two recirculation loop operations. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the reduction factors specified in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore ALPHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.J Local LHGR

During power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed design LHGR.

$$\text{LHGR} \leq \text{LHGRd}$$

LHGRd = Design LHGR

The values for Design LHGR for each fuel type are specified in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS4.5.I Average Planar LHGR

The APLGHR for each type of fuel as a function of average planar exposure shall be checked daily during reactor operation at $\geq 25\%$ rated thermal power.

4.5.J Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq 25\%$ rated thermal power.

PBAPS

LIMITING CONDITIONS FOR OPERATION3.5.J Local LHGR (Cont'd)

If at any time during operation it is determined by normal surveillance that limiting value for LHGR is being exceeded, action shall be initiated within one (1) hour to restore LHGR to within prescribed limits. If the LHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless LHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

1. During power operation the MCPR for the applicable incremental cycle core average exposure and for each type of fuel shall be equal to or greater than the value given in Specification 3.5.K.2 or 3.5.K.3 times K_f , where K_f is as specified in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within one (1) hour to restore MCPR to within prescribed limits. If the MCPR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless MCPR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at >25% rated thermal power.
2. Except as provided in Specification 3.5.K.3, the verification of the applicability of 3.5.K.2.a Operating Limit MCPR Values shall be performed every 120 operating days by scram time testing 19 or more control rods on a rotation basis and performing the following:

- a. The average scram time to the 20% insertion position shall be:
 $\tau_{ave} \leq \tau_B$
- b. The average scram time to the 20% insertion position is determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^n N_i \tau_i}{\sum_{i=1}^n N_i}$$

where: n = number of surveillance tests performed to date in the cycle.

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LIMITING CONDITIONS FOR OPERATION3.5.K Minimum Critical Power Ratio (MCPR) (Cont'd)

2. Except as specified in 3.5.K.3, the Operating Limit MCPR Values shall be as specified in the CORE OPERATING LIMITS REPORT for when

a) requirement 4.5.K.2.a is met, and for when

b) requirement 4.5.K.2.a is not met, where:

$$\tau = \frac{\tau_{ave} - \tau_B}{0.90 - \tau_B}$$

3. If the Surveillance Requirement of Section 4.5.K.2 to scram time test control rods is not performed, the Operating Limit MCPR values shall be as specified in the CORE OPERATING LIMITS REPORT for this condition.

SURVEILLANCE REQUIREMENTS4.5.K Minimum Critical Power Ratio (MCPR) (Cont'd)

N_i = number of active control rods measured in the i th surveillance test.

τ_i = average scram time to the 20% insertion position of all rods measured in the i th surveillance test.

c. The adjusted analysis mean scram time (τ_B) is calculated as follows:

$$\tau_B = \mu + 1.65 \left(\frac{N1}{n \sum_{i=1}^n N_i} \right)^{1/2} \sigma$$

Where:

μ = mean of the distribution for average scram insert time to the 20% position = 0.694 sec.

$N1$ = total number of active control rods measured in specification 4.3.C.1.

σ = standard deviation of the distribution for average scram insertion time to the 20% position = 0.016

PBAPS

Tables 3.5.K.2 and 3.5.K.3 have been removed from former Technical Specification pages 133d and 133e, respectively, and the associated information has been relocated to the Core Operating Limits Report.

PBAPS

3.5 BASES (Continued)H. Engineered Safeguards Compartments Cooling and Ventilation

One unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicated that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured. Ventilation associated with the High Pressure Service Water Pumps is also associated with the Emergency Service Water pumps, and is specified in Specification 3.9.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR Part 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent, secondarily, on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The Technical Specification APLHGR is the LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in the applicable figure for each fuel type in the CORE OPERATING LIMITS REPORT.

Only the most limiting APLHGR operating limits are shown in the figures for the multiple lattice fuel types. Compliance with the lattice-specific APLHGR limits is ensured by using the process computer. When an alternate method to the process computer is required (i.e. hand calculations and/or alternate computer simulation), the most limiting lattice APLHGR limit for each fuel type shall be applied to every lattice of that fuel type.

The calculational procedure used to establish the APLHGR for each fuel type is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (G.E.) calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. A complete discussion of each code employed in the analysis is presented in Reference 4. Input and model changes in the Peach Bottom loss-of-coolant analysis which are different from the previous analyses performed with Reference 4 are described in detail in Reference 8. These changes to the analysis include: (1) consideration of the counter current flow limiting (CCFL) effect, (2) corrected code inputs, and (3) the effect of drilling alternate flow paths in the bundle lower tie plate.

PBAPS

3.5 BASES (Cont'd)J. Local LHGR

This specification assures that the linear heat generation rate in any 8X8 fuel rod is less than the design linear heat generation. The maximum LHGR shall be checked daily during reactor operation at $> 25\%$ power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be at the design LHGR below 25% rated thermal power, the peak local LHGR must be a factor of approximately ten (10) greater than the average LHGR which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and Reference 7. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not violated during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The transients evaluated are as described in Reference 7.

PBAPS

4.5.K Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

4.5.L MCPR Limits for Core Flows Other Than Rated

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves in the CORE OPERATING LIMITS REPORT were developed generically and are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow rate (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of the core flow, divided by the operating limit MCPR determines the K_f .

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

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4.5.L MCPR Limits for Core Flows Other Than Rated (Cont'd.)

The K_f factors specified in the CORE OPERATING LIMITS REPORT are acceptable for Peach Bottom Unit 3 operation because the operating limit MCPR is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f .

PBAPS

The following Figures have been removed from the Technical Specifications and the associated information has been relocated to the Core Operating Limits Report:

Figure 3.5.K.1, former page 142
Figure 3.5.K.2, former page 142a
Figure 3.5.1.E, former page 142d
Figure 3.5.1.F, former page 142e
Figure 3.5.1.G, former page 142f
Figure 3.5.1.H, former page 142g
Figure 3.5.1.I, former page 142h
Figure 3.5.1.J, former page 142i
Figure 3.5.1.K, former page 142j

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6.9.1 Routine Reports (Cont'd)c. Annual Safety/Relief Valve Report

Describe all challenges to the primary coolant system safety and relief valves. Challenges are defined as the automatic opening of the primary coolant safety or relief valves in response to high reactor pressure.

d. Monthly Operating Report

Routine reports of operating statistics and shutdown experience and a narrative summary of the operating experience shall be submitted on a monthly basis to the Office of Management and Program Analysis (or its successor), U.S. Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the appropriate Regional Office, to be submitted no later than the 15th of the month following the calendar month covered by the report.

e. Core Operating Limits Report

(1) Core operating limits shall be established and shall be documented in the CORE OPERATING LIMITS REPORT prior to each Operating Cycle, or prior to any remaining portion of an Operating Cycle, for the following:

- a. The APLHGR for Specification 3.5.I,
- b. The MCPR for Specification 3.5.K,
- c. The K_f core flow adjustment factor for Specification 3.5.K,
- d. The LHGR for Specification 3.5.J,
- e. The upscale flow biased Rod Block Monitor setpoint and the upscale high flow clamped Rod Block monitor setpoint of Specification 3.2.C.

(2) The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents as amended and approved:

- a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version)
- b. Philadelphia Electric Company Methodologies as described in:
 - (1) PECO-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code"

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6.9.1 Routine Reports (Cont'd)

- (2) PECO-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECo)"
 - (3) PECO-FMS-0003-A, "Steady-State Fuel Performance Methods Report"
 - (4) PECO-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis"
 - (5) PECO-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"
- (3) The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.
 - (4) The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be submitted upon issuance for each Operating Cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENT NOS. 154 AND 155 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 March 8, 1990 as supplemented on April 26, 1990, 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 licensee's April 26, 1990 letter proposed editorial and administrative revisions to the TS changes proposed in its original March 8, 1990 submittal. The staff also repaginated portions of the TS without altering the language therein. The staff has determined that these additional changes do not substantially alter the actions noticed or affect the proposed determination that the amendments involve no significant hazards consideration published April 18, 1990. The amendments would modify Technical Specifications (TS) having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications." The proposed amendments also involve miscellaneous administrative changes.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

- (1) The Definition section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specification 3.5.I

The Average Planar Linear Heat Generation Rate (APLHGR) limits for this specification are specified in the COLR.

(b) Specification 3.5.J

The Linear Heat Generation Rate (LHGR) limits for this specification are specified in the COLR.

(c) Specification 3.5.K

The Minimum Critical Power Ratio (MCPR) limits and the MCPR flow adjustment factor (K_f) for this specification are specified in the COLR.

(d) Specification 3.2.C

The upscale high flow clamped Rod Block Monitor (RBM) setpoint, which is also used in the upscale flow-biased RBM setpoint, of this specification is specified in the COLR.

These changes to the specifications also required changes to the TS Bases. Based on our review, we conclude that the changes to these Bases are acceptable.

(3) Specification 6.9.1.e was added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(a) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

(b) Philadelphia Electric Company Methodologies as described in:

1. PECO-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 Using the FIBWR Computer Code."

2. PECO-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECo)."
3. PECO-FMS-0003-A, "Steady-State Fuel Performance Methods Report."
4. PECO-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis."
5. PECO-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis."

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

In response to an April 5, 1990 NRC telecon request, the licensee withdrew Unit 3 page 140b and pages 17, 24, and 140c for both units from the proposed list of changed TS pages in its April 26, 1990 submittal. The staff has reviewed these changes and associated changes which removed all references to topical report PECO-FMS-0006. The staff concludes these changes are acceptable because they remove references to a topical report which has not yet received NRC approval.

The proposed amendments also involved miscellaneous administrative changes to TS Table 3.2.C and various TS pages including those for TS 4.5.K.2.C, Bases for TS 3.5.H, 3.5.I, 3.5.L, and 4.5.L, and TS 6.9.1.C. The staff determined that these changes correct errors, achieve consistency, and provide clarifications, and thus are acceptable.

The revised Technical Specification pages approved and issued by the staff in these amendments differ from the proposed pages in the licensee's submittals to allow for appropriate pagination. Specifically, portions of TS Bases 4.5.L were moved from page 141a to page 141b and portions of

TS 6.9.1.e.(2) were moved from page 256 to page 256a on Units 2 and 3 and portions of the TS List of Figures were moved from page iv to a new page iva on Unit 3. The staff made no changes to the wording in the licensee's proposed TS pages and notified the licensee regarding this reformatting.

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 and changes to the surveillance requirements. 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. The amendments also involve changes to record-keeping and reporting requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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 (55 FR 14517) on April 18, 1990 and consulted with the Commonwealth of Pennsylvania. No public comments were received and the Commonwealth of Pennsylvania did not have any comments.

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.

Dated: May 21, 1990

Principal Contributors:

D. Fieno
T. Dunning