September 1, 1989

Docket No. 50-278

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Mr. George A. Hunger, Jr.
Director-Licensing
Philadelphia Electric Company
Correspondence Control Desk
P. O. Box 7520
Philadelphia, Pennsylvania 19101

Dear Mr. Hunger:

SUBJECT: RELOAD FOR CYCLE 8 OPERATION (TAC NO. 68878)

RE: PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

The Commission has issued the enclosed Amendment No. 150 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Unit No. 3. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 7, 1988.

This amendment includes three categories of changes which involve the operating limits for all fuel types for Cycle 8 operation, the slope of the Average Power Range Monitor scram and rod block setpoints and administrative changes, primarily to the BASES.

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

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 150 to License No. DPR-56
2. Safety Evaluation

cc w/enclosures:
See next page

[PEACH BOTTOM LTR]

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	REMartin:cdd		for WButler
8/1/89	8/10/89	8/21/89	9/1/89

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

September 1, 1989

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P.O. Box 7520
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Sincerely,

Robert Martin

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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:

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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. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
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 July 7, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), 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 amendment to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 150 , are hereby incorporated in the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/S/ Mohan C. Thadani for

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 1, 1989

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
M'OBrien	REMartin:cdd	MYoung	for WButler
8/1/89	8/10/89	8/21/89	/ /89

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 1, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 150

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

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

Remove	Pages
iv	iv
1	1
9	9
9a	9a
10	10
11	11
11a	11a
13	13
15	15
16	16
17	17
18	18
33	33
37	37
40	40
73	73
74	74
133a	133a
133b	133b
133c	133c
133d	133d
133e	133e
140	140
140b	140b
140c	140c
140d	-
142	142
142a	142a
-	142e
-	142f

PBAPS

LIST OF FIGURES

Amendment No. ~~14~~, ~~41~~, ~~45~~, ~~46~~, ~~62~~, ~~79~~, ~~92~~, ~~104~~, ~~107~~, ~~114~~, ~~126~~, ~~142~~, 150

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.

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.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objectives:

The objective of the Safety Limits is to establish limits which assure the integrity of the fuel cladding.

Specification:

A. Reactor Pressure $\geq$ 800 psia and Core Flow $\geq$ 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.04 for two recirculation loop operation, or 1.05 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objectives:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.58W + 62\% - 0.58\,\Delta W$

where:

S = Setting in percent of rated thermal power (3293 MWt)

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

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

ΔW = 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 scram to preserve the original (two loop) relationship between APRM scram setpoint and recirculation drive flow or by adjusting the APRM flux trip setting. $\Delta W = 0$ for two loop operation.

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 13.4 KW/ft for BP/P8X8R and LTA fuel and 14.4 KW/ft for GE8X8EB 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.

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.

SAFETY LIMIT | LIMITING SAFETY SYSTEM SETTING

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

When the reactor pressure is $\leq$ 800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

B. APRM Rod Block Trip Setting

SRB $\leq$ (0.58 W + 50% - 0.58 Δ W)

where:

SRB = Rod block setting in percent of rated thermal power (3293 MWt)

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

Δ W = Difference between two loop and single loop effective recirculation drive flow at the same core flow. During single loop operation, the reduction in trip setting (-0.58 Δ W) is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between APRM Rod block setpoint and recirculation drive flow or by adjusting the APRM Rod block trip setting. Δ W = 0 for two loop operation.

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

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

B. APRM Rod Block Trip Setting

$$SRB \leq (0.58\ W + 50\% - 0.58\Delta W)\ \frac{FRP}{MFLPD}$$

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 BP/P8X8R and LTA fuel and 14.4 KW/ft for GE8X8EB 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)

Amendment No. ~~77~~, ~~79~~, ~~115~~, 150

1.1 BASES: FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure ≥ 800 psia and Core Flow ≥ 10% of Rated

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis described in references 1 and 3 for two recirculation loop operation. The Safety Limit MCPR is increased by 0.01 for single-loop operation as discussed in reference 4.

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 annunciation 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.

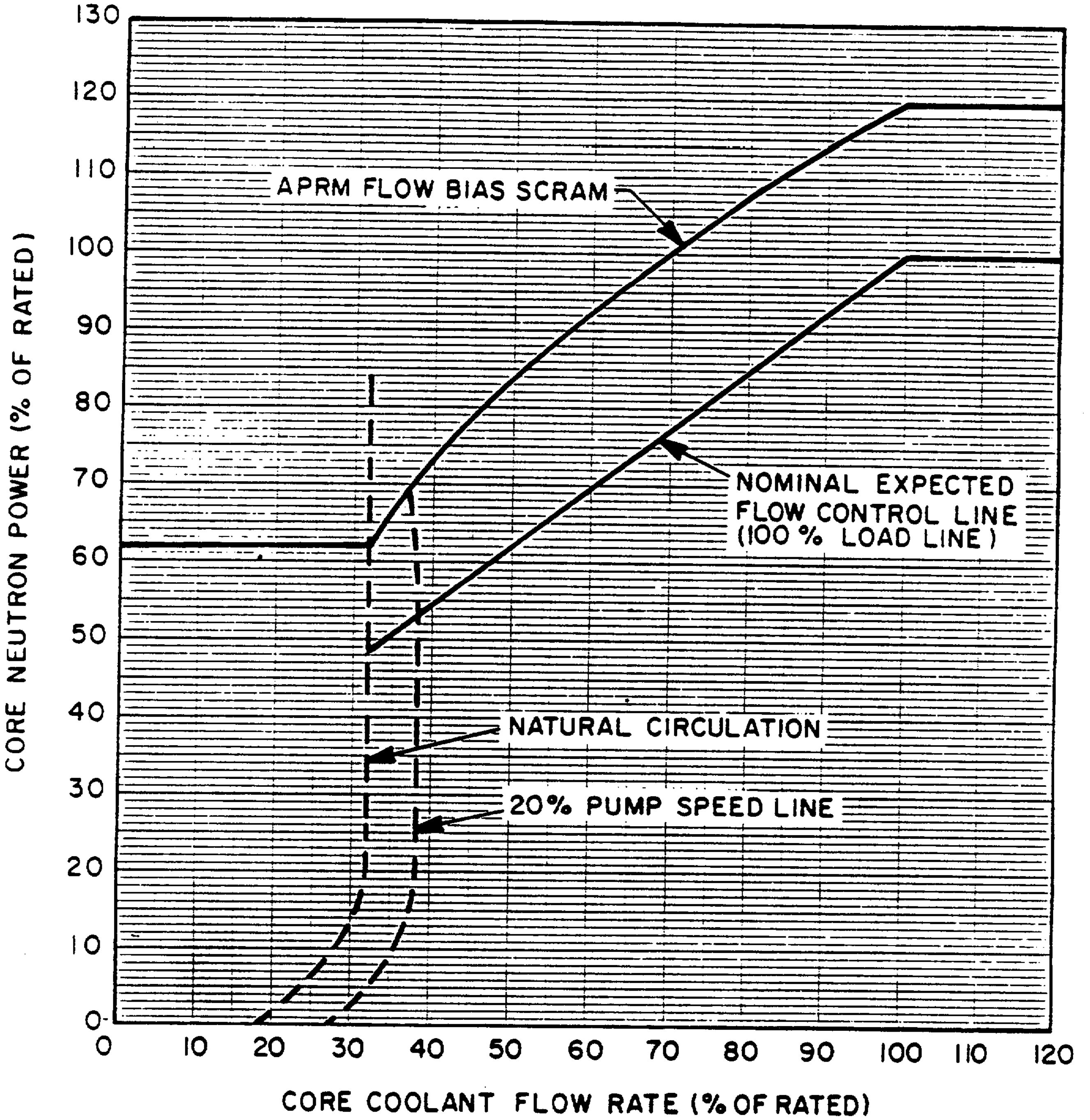
E. References

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 Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

4. "Peach Bottom Atomic Power Station Units 2 and 3 Single-Loop Operation", NEDO-24229-1, May 1980.



APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS

FIGURE 1.1-1

2.1 BASES: FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Peach Bottom Atomic Power Station Units have been analyzed throughout the spectrum of planned operating conditions up to or above the thermal power condition required by Regulatory Guide 1.49. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7.1 of the FSAR. In addition, 3293 MWt is the licensed maximum power level of each Peach Bottom Atomic Power Station Unit, and this represents the maximum steady state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. Conservatism incorporated into the transient analyses is documented in NEDE-24011-P-A (as amended).

2.1 BASES (Cont'd)

For analyses of the thermal consequences of the transients, a MCPR equal to or greater than the operating limit MCPR given in Specification 3.5.K is conservatively assumed to exist prior to initiation of the limiting transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady state operation without forced recirculation will not be permitted. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculating pumps.

In summary:

i. The abnormal operational transients were analyzed at or above the maximum power level required by Regulatory Guide 1.49 to determine operating limit MCPR's.

ii. The licensed maximum power level is 3293 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual trip settings are discussed in the following paragraphs.

A. Neutron Flux Scram

The Average Power Range Monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (3293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The pressure relief system for each unit at the Peach Bottom Atomic Power Station has been sized to meet two design bases. First, the total capacity of the safety/relief valves and safety valves has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4.1 of subsection 4.4 of the FSAR which states that the nuclear system safety/relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which show compliance with the ASME Code requirements are presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report submitted in Appendix K.

Eleven safety/relief valves and two safety valves have been installed on Peach Bottom Units 2 and 3. The analysis of the worst overpressure transient is provided in the Supplemental Reload Licensing Submittal and demonstrates margin to the code allowable overpressure limit of 1375 psig.

The safety/relief valve settings satisfy the Code requirements that the lowest valve setpoint be at or below the vessel design pressure of 1250 psig. These settings are also sufficiently above the normal operating pressure range to prevent unnecessary cycling caused by minor transients.

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome less than 75 psig.

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Item	Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (7)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
1	1	Mode Switch In Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
2	1	Manual Scram		X	X	X	2 Instrument Channels	A
3	3	IRM High Flux	≤120/125 of Full Scale	X	X	(5)	8 Instrument Channels	A
4	3	IRM Inoperative		X	X	(5)	8 Instrument Channels	A
5	2	APRM High Flux	(0.58W+62-0.58ΔW) FRP/MFLPD (12) (13)			X	6 Instrument Channels	A or B
6	2	APRM Inoperative	(11)	X	X	X	6 Instrument Channels	A or B
7	2	APRM Downscale	≥2.5 Indicated on Scale			(10)	6 Instrument Channels	A or B
8	2	APRM High Flux in Startup	≤15% Power	X	X		6 Instrument Channels	A
9	2	High Reactor Pressure	≤1055 psig	X(9)	X	X	4 Instrument Channels	A
10	2	High Drywell Pressure	≤2 psig	X(8)	X(8)	X	4 Instrument Channels	A
11	2	Reactor Low Water Level	≥0 in. Indicated Level	X	X	X	4 Instrument Channels	A

NOTES FOR TABLE 3.1.1 (Cont'd)

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

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

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

    FRP = fraction of rated thermal power (3293 MWt).
    MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for BP/P8X8R and LTA fuel and 14.4 KW/ft for GE8X8EB fuel.

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

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

    Delta W = The difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.58 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	APRM Upscale (Flow Biased)	$\leq(0.58W+50-0.58\Delta W) \times \frac{FRP}{MFLPD}$ (2)	6 Inst. Channels	(10)
4	APRM Upscale (Startup Mode)	$\leq$12%	6 Inst. Channels	(10)
4	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(10)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq(0.66w+41-0.66\Delta W) \times \frac{FRP}{MFLPD}$ (2) with a maximum of $\leq$ 107%	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$2.5 indicated on scale	2 Inst. Channels	(1)
6	IRM Downscale (3)	$\geq$2.5 indicated on scale	8 Inst. Channels	(10)
6	IRM Detector not in Startup Position	(8)	8 Inst. Channels	(10)
6	IRM Upscale	$\leq$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	$\leq$25 gallons	1 Inst. Channel	(9)

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

   FRP = fraction of rated thermal power (3293 MWt)

   MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for BP/P8X8R and LTA fuel and 14.4 KW/ft for GE8X8EB fuel.

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

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

   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 $\geq$ 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.

LIMITING CONDITIONS FOR OPERATION

3.5.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 approved 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) shown in the applicable figures for BP/P8X8R, LTA and GE8X8EB fuel types during two recirculation loop operations. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the following reduction factors: 0.81 for BP/P8X8R and LTA fuel and 0.73 for GE8X8EB fuel. 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.

SURVEILLANCE REQUIREMENTS

4.5.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 ≥ 25% rated thermal power.

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.

LHGR ≤ LHGRd

LHGRd = Design LHGR
13.4 KW/ft for BP/P8X8R and LTA fuel
14.4 KW/ft for GE8X8EB fuel

4.5.J Local LHGR

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

LIMITING CONDITIONS FOR OPERATION

3.5.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 Kf, where Kf is as shown in Figure 3.5.1.E. 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 REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at $\geq 25\%$ 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.

LIMITING CONDITIONS FOR OPERATION

3.5.K Minimum Critical Power Ratio(MCPR) (Cont'd)

2. Except as specified in 3.5.K.3, the Operating Limit MCPR Values are as follows:

   a. If requirement 4.5.K.2.a is met:
   The Operating Limit MCPR values are as given in Table 3.5.K.2

   b. If requirement 4.5.K.2.a is not met:
   The Operating Limit MCPR values as a function of τ are as given in Figures 3.5.K.1 and 3.5.K.2

Where:

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

3. The Operating Limit MCPR values shall be as given in Table 3.5.K.3 if the Surveillance Requirement of Section 4.5.K.2 to scram time test control rods is not performed.

SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio(MCPR) (Cont'd)

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

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

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

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

Where:

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

N_1 = 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

Table 3.5.K.2

OPERATING LIMIT MCPR VALUES
FOR VARIOUS CORE EXPOSURES*

Fuel Type	MCPR Operating Limit For Incremental Cycle Core Average Exposure**	
	BOC to 2000 MWD/t Before EOC	2000 MWD/t before EOC To EOC
Standard Operating Conditions		
BP/P8X8R	1.21	1.26
LTA	1.21	1.26
GE8X8EB	1.21	1.26
Increased Core Flow		
BP/P8X8R	1.21	1.27
LTA	1.21	1.27
GE8X8EB	1.21	1.27

* If requirement 4.5.K.2.a is met.

** These values shall be increased by 0.01 for single loop operation.

Table 3.5.K.3

OPERATING LIMIT MCPR VALUES
FOR VARIOUS CORE EXPOSURES*

Fuel Type	MCPR Operating Limit For Incremental Cycle Core Average Exposure**	
	BOC to 2000 MWD/t Before EOC	2000 MWD/t before EOC To EOC
Standard Operating Conditions		
BP/P8X8R	1.26	1.30
LTA	1.26	1.30
GE8X8EB	1.26	1.30
Increased Core Flow		
BP/P8X8R	1.26	1.31
LTA	1.26	1.31
GE8X8EB	1.26	1.31

* If surveillance requirement of section 4.5.K.2 is not performed.

** These values shall be increased by 0.01 for single loop operation.

3.5 BASES (Continued)

H. Engineering 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 figures for each fuel type.

Only the most limiting and least limiting APLHGR operating limits are shown in the figures for the multiple lattice fuel types. Compliance with the lattice-specific, approved 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.

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 action 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.

3.5.L. BASES (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominately occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss-of-Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate.

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7 and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE 20566 (Draft), August 1974.

5. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter, G. L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

6. DELETED.

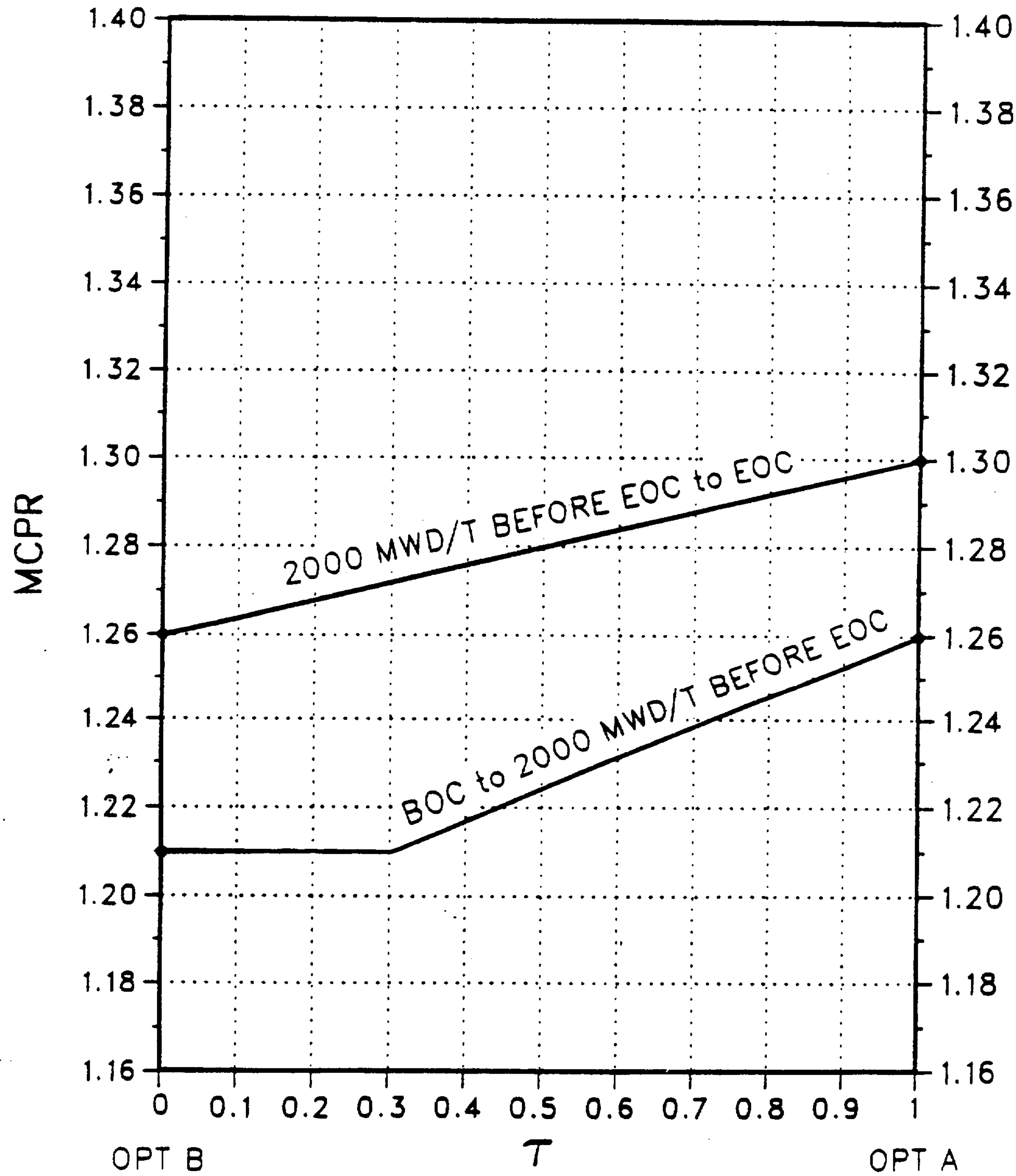
7. "General Electric Standard Application for Reactor Fuel", NEDO-24011-P-A (as amended).

8. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, NEDO-24081, December 1977, and for Unit 3, NEDO-24082, December 1977.

9. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, Supplement 1, NEDE-24081-P, November 1986, and for Unit 3, NEDE-24082-P, December 1987.

FIGURE 3.5.K.1

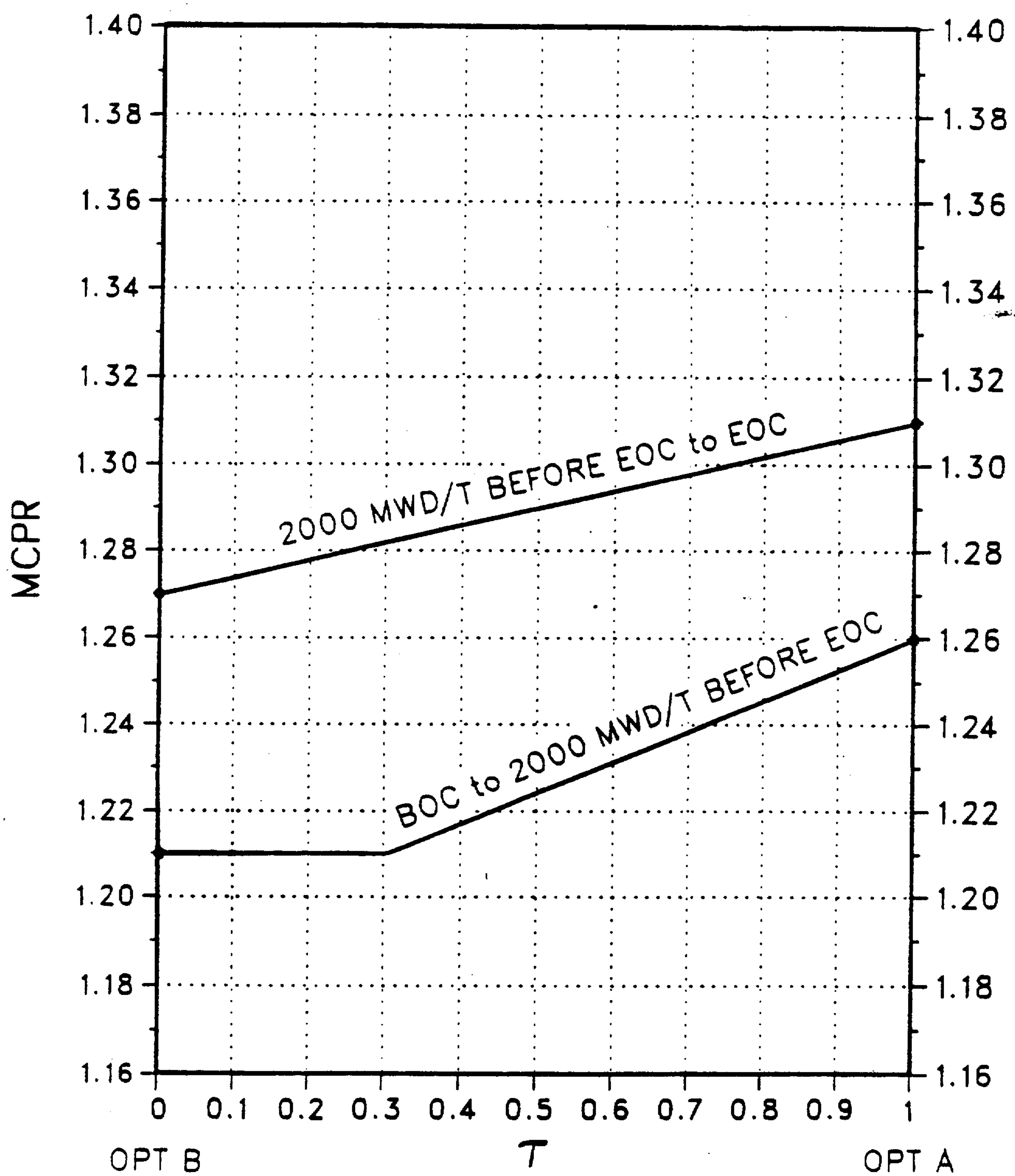
MCPR OPERATING LIMIT vs τ
FUEL TYPES: BP/P8X8R,LTA,GE8X8EB
(STANDARD OPERATING CONDITIONS)



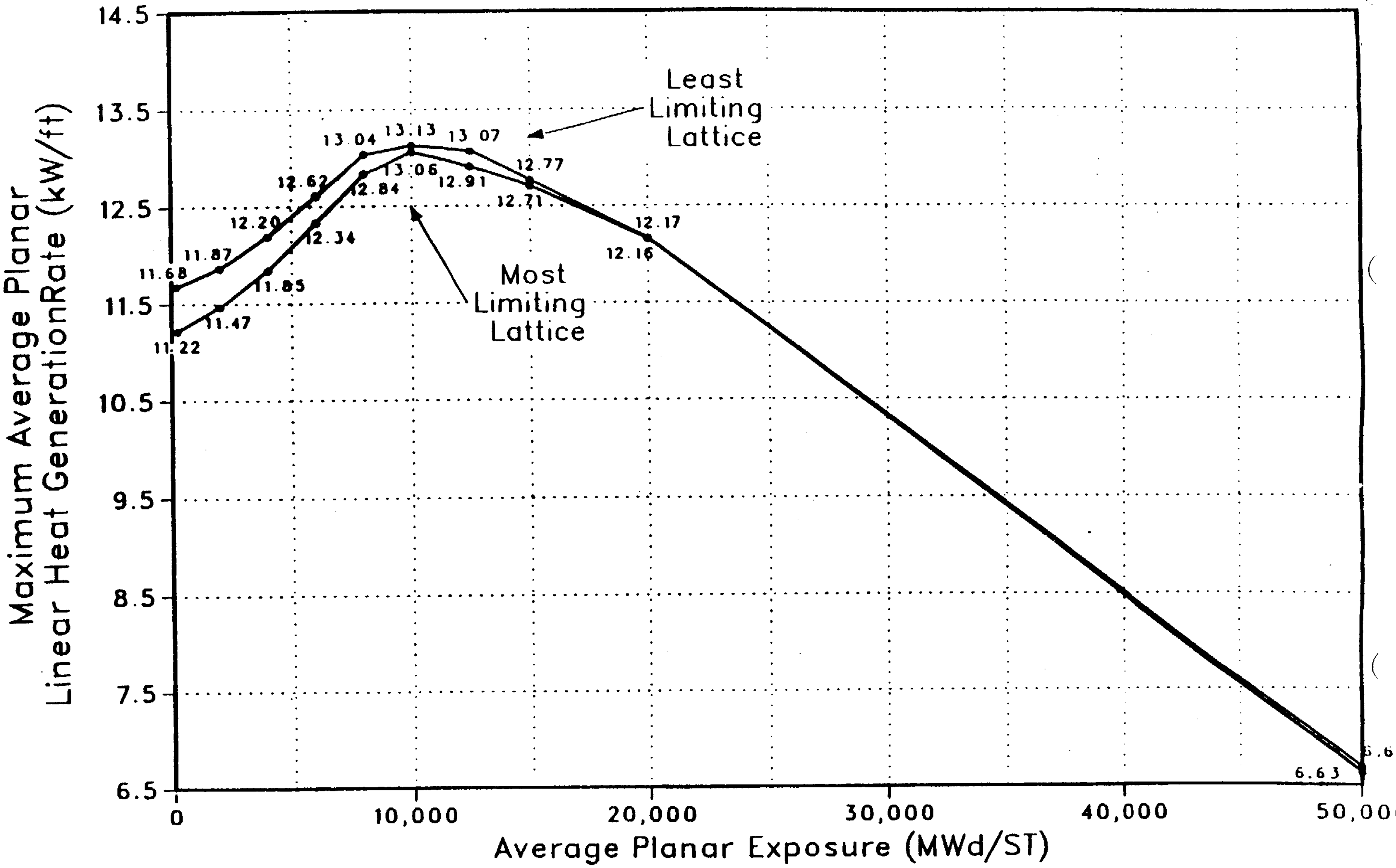
Amendment No. ~~41~~, ~~79~~, ~~85~~, ~~92~~,
~~114~~, 150

FIGURE 3.5.K.2

MCPR OPERATING LIMIT vs τ
FUEL TYPES: BP/P8X8R,LTA,GE8X8EB
(INCREASED CORE FLOW)

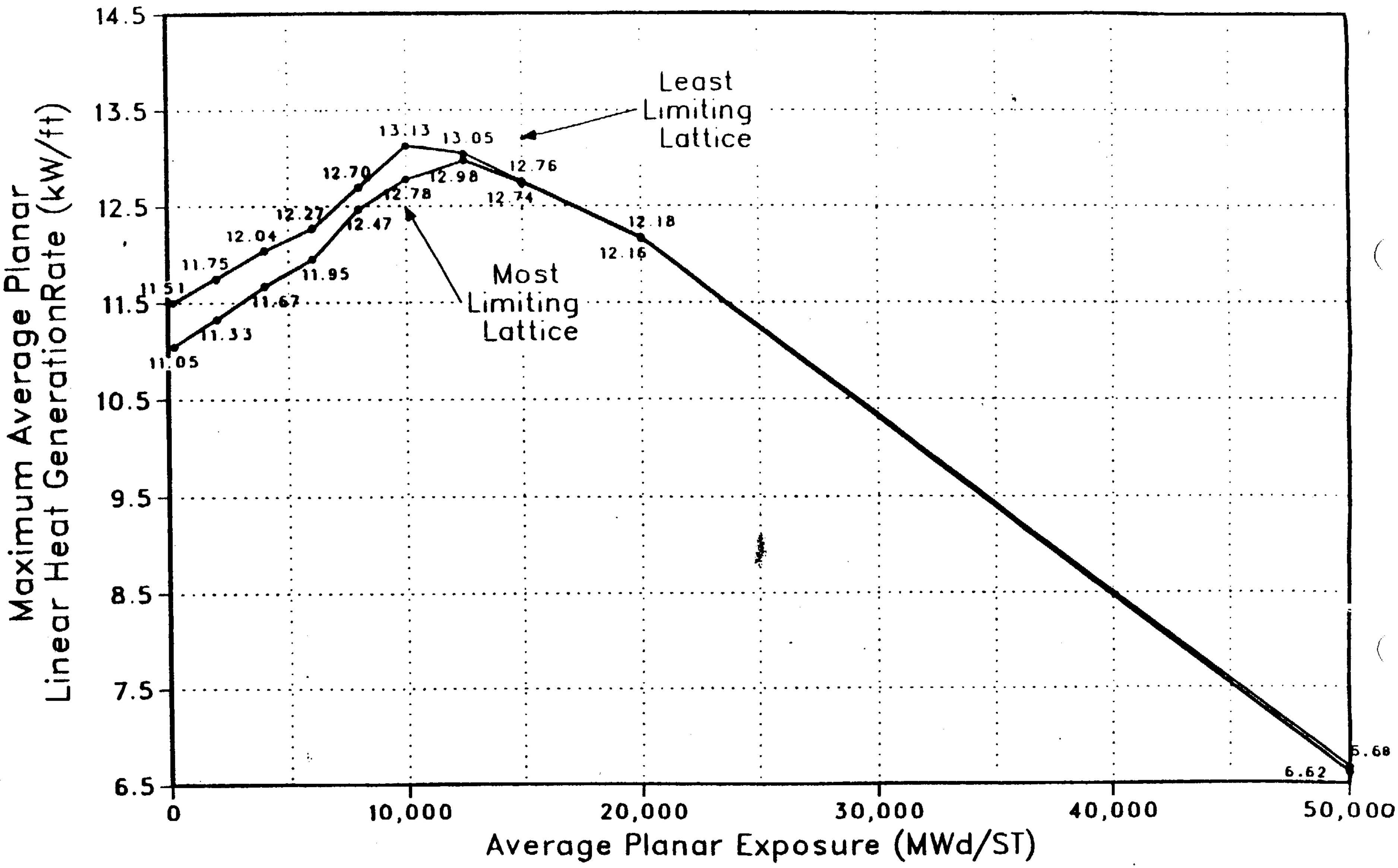


Amendment NO. ~~41~~, ~~79~~, ~~85~~, ~~92~~, ~~107~~, ~~114~~, 150



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
FOR FUEL TYPE BD319A (GE8X8EB)

FIGURE 3.5.1F



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
FOR FUEL TYPE BD321A (GE8X8EB)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENT NO. 150 TO FACILITY OPERATING LICENSES NO. 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 NO. 3

DOCKET NO. 50-278

1.0 INTRODUCTION

By letter dated July 7, 1988, Philadelphia Electric Company requested an amendment to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Unit No. 3. This amendment would revise the Technical Specifications (TS) to: (1) incorporate the operating limits for all fuel types for Cycle 8 operation, (2) incorporate a change in slope of the flow biased Average Power Range Monitor (APRM) scram and rod block setpoints to provide increased operating flexibility during power ascension, (3) correct a typographical error, (4) clarify a definition of Average Planar Linear Heat Generation Rate (APLHGR), and (5) make various changes to the Bases resulting from the core reload. TS changes were proposed for the operation of Peach Bottom Atomic Power Station, Unit No. 3 for Cycle 8 (PB3C8) with a reload using General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies. Enclosed were the requested TS changes and reports (Refs. 2 and 3) discussing the reload and analyses done to support and justify Cycle 8 operation and extended power-flow operating regions.

The reload for Cycle 8 is generally a normal reload with no unusual core features or characteristics. TS changes are primarily related to Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) limits for the new fuel and Minimum Critical Power Ratio (MCPR) limits for all of the fuel using Cycle 8 core and transient parameters. The new fuel is the GE extended burnup fuel which is extensively used in GE fueled core reloads.

The submittal also proposes extensions of the standard allowed operating regions on the reactor temperature and power-flow map. The Extended Load Line Limit Analysis (ELLLA), Increased Core Flow (ICF), and the Final Feedwater Temperature Reduction (FFWTR) proposed modes of extended operation are similar to those approved on a number of other BWRs in recent years. Except for changes to the flow-biased APRM scram and rod block setpoints for ELLLA and some additional MCPR limits for ICF, they require no other changes to Cycle 8 TS.

As noted above, the supporting analysis was performed by GE for PECO using NRC approved methods and codes. In the subject Peach Bottom, Unit 3 submittal, as well as in the previous PECO submittal for Peach Bottom, Unit 2, Cycle 8, GE has proposed a new approach of only including the curves for the most limiting and least limiting MAPLHGRs as a function of planar average exposure values for each fuel type in the TS. During power operation, the process computer would check that the APLHGR for each type of fuel as a function of axial location and average planar exposure was within the limits based on the applicable APLHGR limit values which had been approved for the respective axial lattice of a given fuel type. When the process computer is not available and hand calculations are required, the most limiting lattice APLHGR limits for each fuel type will apply to every lattice of that fuel type. The lattice dependent MAPLHGR values are provided in Reference 3. This procedure for treating lattice-dependent MAPLHGRs for a given fuel type has been reviewed and approved by the staff and provided in Reference 4.

2.0 EVALUATION

2.1 Reload Description

The PB3C8 reload will retain 284 P8X8R, 284 BP8X8R, and 4 Lead Test Assemblies (LTAs) from the previous cycle and add 192 new GE8X8EB fuel assemblies. All of the fuel assemblies are GE manufactured. The reload is based on a previous cycle core nominal average exposure of 15.9 GWd/ST and Cycle 8 end of cycle exposure of 18.9 GWd/ST. The loading will be a conventional scatter pattern with low reactivity fuel on the core periphery.

2.2 Fuel Design

The new fuel for Cycle 8 is the GE extended burnup fuel GE8X8EB. The fuel designations are BD319A and BD321A. This fuel type has been approved in the Safety Evaluation Report for Amendment 10 to GESTAR II (Refs. 5 and 6). The specific descriptions of this fuel are presented for PB3C8 in Reference 3. These fuel descriptions are acceptable.

In operation the GE8X8EB fuel will be assigned a number of lattice regions and appropriate MAPLHGR limits, which have been determined by approved thermal-mechanical and loss-of-coolant accident (LOCA) analyses, will be applied to each of these regions. The process computer contains, and acts on, full details of the MAPLHGR information. The TS present the least and most limiting lattice MAPLHGRs as a function of average planar exposure. When hand calculations of MAPLHGRs are required (process computer inoperative), the most limiting MAPLHGR values are used for all lattices of a given fuel type. These TS are acceptable. A proprietary report, reviewed by the NRC staff and, available to the Peach Bottom, Unit 3 staff, provides complete details of the lattice definitions and MAPLHGR limits.

The proposed LHGR limit for the GE8X8EB fuel is 14.4 KW/ft rather than the 13.4 KW/ft for the other GE fuel in PB3C8. This LHGR limit has been reviewed and accepted for this fuel in the GE extended burnup fuel review (Ref. 5). (See the referrals in Reference 5 to References 18 and 19. These references are responses to questions and presentations relating to the GE8X8EB fuel which provide information on the 14.4 KW/ft LHGR.) This LHGR limit is acceptable for the GE8X8EB fuel in PB3C8.

2.3 Nuclear Design

The nuclear design for PB3C8 has been performed by GE with the approved methodology described in GESTAR-II (Ref. 6). The results of these analyses are given in the reload report (Ref. 2) in standard GESTAR-II format. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 2.2% and 1.3% delta K at beginning-of-cycle (BOC) and at the exposure of minimum shutdown margin, respectively, thus fully meeting the required 0.38% delta K. The Standby Liquid Control System also meets shutdown requirements with a shutdown margin of 4.2% delta K. Because these and other PB3C8 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design for PB3C8 has been performed by GE with the approved methodology described in GESTAR-II (Ref. 6) and the results are given in the reload report (Ref. 2). The parameters used for the analysis are those approved in Reference 6 for the Peach Bottom BWR-4 class except for the parameters listed in Appendix C of Reference 2. The GEMINI system of methods (approved in Ref. 7) was used for the relevant transient analyses.

The Operating Limit MCPR (OLMCPR) values are determined by the limiting transients, which are usually Rod Withdrawal Error (RWE), Feedwater Controller Failure (FWCF), Loss of Feedwater Heating (LFH), and Load Rejection Without Bypass (LRWBP). The analyses of these events for PB3C8, using the ODYN Option A and B approaches for pressurization transients, provide new Cycle 8 TS values of the OLMCPR as a function of average scram time, for operation in both standard and extended operating regions.

For PB3C8 PECO has elected, following standard practice, to have exposure dependent OLMCPRs. Two exposure regions from BOC to end-of-cycle (EOC) were analyzed: (1) BOC to EOC - 2 GWd/ST and (2) EOC - 2 GWd/ST to EOC. For standard operating conditions, the LRWBP event is controlling at both Option A and B limits except at BOC Option B where the RWE event is controlling. These OLMCPR results are reflected in TS changes. Approved methods (Ref. 6) were used to analyze these events (and others which could be limiting) and the analyses and results are acceptable and fall within expected ranges.

The Safety Limit MCPR (SLMCPR) is set so that less than 0.1 percent of the fuel pins in the core are subject to boiling transition when some fuel in the core is at the SLMCPR. The SLMCPR is being changed to 1.04 (1.05 for single recirculation loop operation) for PB3C8. This change has been approved by the NRC for D-lattice cores operating with the second successive reload core of high bundle R-factor fuel types (Ref. 10). The licensee states that all fuel types to be loaded for Cycle 8 operation (that is, BP/P8X8R, LTA, and GE8X8EB) are high bundle R-factor fuel types. Thus the change to the SLMCPR is acceptable.

The mean and standard deviation of the control rod scram speed data that are used to compute the adjusted mean scram time (Tau) are being changed. These new values of the mean and standard deviation results in a calculation of a small and more restrictive value of Tau. These changes lead to a conservative determination of Tau and OLMCPR and are therefore acceptable.

The Peach Bottom 3 TS have staff approved provisions for incore neutron detector monitoring of thermal-hydraulic stability according to the recommendation of GE SIL-380. Thus cycle specific stability calculations are not required, either for standard conditions or the extended temperature and power-flow conditions proposed for Cycle 8 operation (see Section 2.6).

2.5 Transient and Accident Analyses

The transient and accident analysis methodologies used for PB3C8 are described and NRC approval indicated in GESTAR II (Ref. 6). The GEMINI system of methods (Ref. 7) option was used for transient analyses. The limiting MCPR events for PB3C8 are indicated in Sections 2.4 and 2.6. The core wide transient analysis methodologies are acceptable and fall within expected ranges.

The RWE was analyzed on a plant and cycle specific basis (as opposed to the statistical approach) and a rod block monitor setpoint of 107 was selected to provide an OLMCPR of 1.21 for all fuel types. The mislocated assembly event is not analyzed for reload cores on the basis of (NRC approved, see Ref. S.2-59 of Ref. 6) studies indicating the small probability of an event exceeding MCPR limits. The misorientation event was analyzed with standard methods for the PB3C8 D lattice fuel, giving a nonlimiting MCPR of 1.18. The local transient event analyses are thus acceptable.

The limiting pressurization event for establishing overpressure protection margin, the main steam isolation valve closure event with flux scram, was analyzed with standard GESTAR II methods and gave results for peak steam dome and vessel pressures well under required limits. These are acceptable methodologies and results.

LOCA analysis, using approved methodologies (SAFE/REFLOOD/CHASTE) and parameters, were performed to provide MAPLHGR values for the new reload fuel assemblies (GE8X8EB). These analyses and results are acceptable.

Since some parameters of the generic rod drop accident (RDA) were not bounding for PB3C8, cycle specific RDA analyses were done for cold and hot shutdown conditions. These were done with standard, approved GE methods. The results were well within the required 280 cal/gm limit. The analyses and results are acceptable.

2.6 Changes to APRM and RBM Setpoints

The PB3C8 reload submittal proposes extensions to standard operating regions in the GESTAR II category of "Operating Flexibility Options." The selected options are ELLA, ICF, and FFWTR. These have become commonly selected and approved options for a number of reactors in recent years. These options are described and discussed in appendices A and B of the reload report (Ref. 2) and in GE topical reports (Refs. 8 and 9). These appendices provide the results of transient analyses for setting MCPR limits for Cycle 8 and the topical reports provide generic analyses of transients and accidents, applicable for follow-on cycles as well.

The proposed ELLLA changes the APRM flow-biased scram and rod block lines on the power-flow map, and permits operation up to the new APRM flow-biased rod block line (0.58W + 50%) up to the intersection with the 100 percent power line at a core flow of 87%. In the above, W is the recirculation drive flow in percent of rated. Similarly, the new APRM flow-biased scram line is given as (0.58W + 62%). These are standard changes for ELLA. For ICF, the proposed flow increase is to 105 percent core flow at 100 percent power. The increased flow would be allowed throughout the cycle and after normal EOC 8 (with or without FFWTR) with reactivity coast down to 70 percent power. After EOC 8, the ICF would be bounded by 110 percent of rated flow. The proposed FFWTR involves valving out last stage feedwater heaters (going to a feedwater temperature of about 328°F) and is proposed only for operation after normal EOC 8.

For the ELLLA extension, the topical report (Ref. 9) discusses a full range of transient and accident events relevant to the region extension, and presents results of calculations or previously approved conclusions. In addition, Appendix B of the PB3C8 reload report (Ref. 2) presents additional calculations of limiting MCPR transients specifically for PB3C8. The transient analyses demonstrate that for reactors such as Peach Bottom 3 which do not have Recirculation Pump Trip for transient scram response assistance, the licensing basis results (e.g., 100 percent flow, 100 percent power for pressurization transients) bound the ELLLA region results (e.g., 87 percent flow, 100 percent power). These conclusions apply to all relevant MCPR events such as pressurization, rod withdrawal, and slow flow runout events. Changes to MCPR TS are not required because of adoption of the ELLLA option. Other relevant areas such as overpressure protection, LOCA, and containment analyses have also been examined by the licensee, and the analyses indicate that results are within allowable design limits. Thermal-hydraulic stability will be provided for by appropriate surveillance. The analyses have been done with approved methodologies and the results are similar to previously approved ELLLA extensions. Thus operation within the ELLLA region is acceptable for PB3C8.

For the ICF and FFWTR extensions, similar to the ELLLA presentation, the topical report (Ref. 8) discusses a full range of relevant transient and accident events and other potential problem areas, and Appendix A of the reload report (Ref. 2) provides analyses of limiting MCPR events for PB3C8. Unlike the situation for ELLLA, the analysis of MCPR events leads, in some situations, to more restrictive MCPR limits, which are cycle dependent. Appendix A presents the results of calculations, using standard methodology, for the most limiting events at the most limiting combination of ICF and/or FFWTR conditions for PB3C8. These are presented for Options A and B and for both exposure ranges considered for standard operating conditions (FFWTR is allowed only for EOC). The results are reflected in the TS which are changed to provide a new MCPR limit for EOC-2000 GWD/ST to EOC operation with ICF. FFWTR operation, within the bounds to be used, is not limiting with or without ICF, compared to standard operating conditions. The RWE results for the standard operation region are not affected with the RBM clipped at 107 in the ICF region. The MCPR analyses for ICF and FFWTR extensions use standard methods, follow previously approved trends, and are acceptable.

The licensee has also evaluated other events and affected system components related to these extensions. These include overpressure protection, fuel loading error, rod drop accident, and LOCA events, none of which are significantly affected by the extensions. As in the case for ELLLA, the thermal-hydraulic stability will be appropriately monitored using GE SIL-380 surveillance, and will thus present no new problem. The licensee has evaluated the effects of ICF induced increased pressure differentials and vibration response on reactor internals, fuel channels, and fuel bundles, and has determined that design limits will not be exceeded. The containment LOCA response was analyzed and the results show no significant impact of ICF and FFWTR. The feedwater nozzle and sparger fatigue usage factors were examined for the effects of extreme programs of FFWTR and EOC power coast down. The analysis leads to the conclusion that there is no significant impact beyond a slightly increased nozzle refurbishment schedule (based on monitored seal leakage). The review of these various licensee evaluations leads to the conclusion that suitable analyses were performed and the results are compatible with other reviews and are acceptable for Peach Bottom 3.

2.7 Techncial Specifications

2.7.1

The TS changes for PB3C8 associated with the reload and operating flexibility options are primarily to provide for:

(a) The new ELLLA APRM flow-biased scram and rod block setpoints. The changes are to TS 2.1.A.1, TS 2.1.B, Figure 1.1-1, Table 3.1-1, and Table 3.2.C and are acceptable.

(b) The new MCPR limits for Cycle 8 and ICF operation. The changes are to TS 3/4.5.K, Tables 3.5.K.2 and 3.5.K.3, and Figures 3.5.K.1 and 3.5.K.2 and are acceptable.

(c) The 14.4 KW/ft LHGR limit for the new GE8X8EB fuel. The changes are to TS 2.1.A, TS 2.1.B, TS 3.5.J, Table 3.1.1 and Table 3.2.C and are acceptable.

(d) MAPLHGR limits for the new fuel. The changes are to TS 3.5.I, Figure 3.5.1.f and Figure 3.5.1.G and are acceptable.

2.7.2

One typographical error is being corrected by the amendment as follows:

(a) TS 3.5.K is incorrectly identified on page 133b as TS 3.5.K.1.

This change is a correction to a TS identification and is acceptable.

2.7.3

Page IV of the list of figures in the Table of Contents is being revised to reflect the MCPR and MAPLHGR figures changed by this amendment.

2.7.4

In the Definitions Section of the TS, a paragraph is being added to define "Average Planar Linear Heat Generation Rate." This is a desirable addition because the term is not defined in the current TS.

2.7.5

The bases for TS 1.1 on pages 13 and 15 are being modified to include a reference for single-loop operation for Peach Bottom Units 2 and 3. This change is acceptable because it provides an appropriate reference for single-loop operation.

2.7.6

The Bases for TS 3.5.I are being modified to incorporate a new paragraph regarding APLHGR operating limits for multiple-lattice fuel bundles. This change is acceptable.

2.7.8

The Bases for the core thermal-hydraulic and physics analyses reference the GE topical report "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A in a number of sections. This topical report is subject to periodic revision by GE. Changes to the topical report have to be approved by the NRC. Wherever this is referenced in the Bases the words "as amended" are appended to refer to the most recently approved version. Bases pages affected are 15, 17, and 140c. There are also changes to the Bases in Sections 2.1 and 2.2 to eliminate information that is redundant to the topical report. These changes will eliminate the need to change the date on the references if the referenced topical report is revised. This approach has been adopted by most licensees and is endorsed by the staff.

2.7.9

A sentence is being added in two places to the Bases for TS 2.1 on pages 17 and 18 that states that abnormal operational transients were analyzed at or above the maximum power level required by Regulatory Guide 1.49 to determine operating limit MCPRs. Reference to a specific power level (that is, 3440 MWt) is being deleted. The change reflects the analyses performed for the core reload.

2.8 Conclusions

We have reviewed the report submitted for the Cycle 8 operation of Peach Bottom 3 with extended operating regions. Based on this review we conclude that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitability reflect the necessary modifications for operation in this cycle.

2.9 References

1. Letter from E. J. Bradley (PECO) to T. M. Murley (NRC), dated July 7, 1988.

2. GE Report 23A5889 Rev. 0, dated January 1988, "Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station, Unit 3, Reload 7, Cycle 8."

3. NEDE-24082-P-1, Supplement 1, Revision 1, January 1988, "Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station, Unit 3" (and Errata and Addenda Sheet No. 7).

4. Letter from J. S. Charnley (GE) to M. W. Hodges (NRC), dated March 4, 1987, "Recommended MAPLHGR Technical Specifications for Multiple Lattice Fuel Designs."

5. Letter (and attachment) from C. Thomas (NRC) to J. Charnley (GE), dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A-6, Amendment 10."

6. GESTAR-II, NEDE-24011, Revision 8, "General Electric Standard Application for Reactor Fuel."

7. Letter (and attachment) from G. Lainas (NRC) to J. Charnley (GE), dated March 22, 1986, "Acceptance for Referencing of Licensing Topical Report NEDE-20411-P-A, 'GE Generic Licensing Reload Report,' Supplement to Amendment 11."

8. "Safety Review of Peach Bottom Atomic Power Station Unit No. 3 at Core Flow Conditions Above Rated Flow Throughout Cycle 6," NEDC-30519, March 1984.

9. "General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Peach Bottom Unit 2, Cycle 7, and Peach Bottom Unit 3, Cycle 7," NEDC-31298, May 1986.

10. Letter from Ashok C. Thadani (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Amendment 14 to General Electric Licensing Topical Report NEDE-24011-P-A, 'General Electric Standard Application for Reactor Fuel,'" dated December 27, 1987.

In addition, typographical corrections were made to the licensee's incoming technical specification pages and bars were added to show the changes.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves 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 amendment involves 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 27234) on June 28, 1989 and consulted with the State of Pennsylvania. No public comments were received and the State 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: September 1, 1989