LIST OF FIGURES

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	LIMITING SAFETY SYSTEM SETTING
1.1 FUEL CLADDING INTEGRITY	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

$SRB \leq (0.58\ W + 50\% - 0.58\ \Delta W)$

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.

ΔW = 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 ($-0.58\ \Delta W$) 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. $\Delta W = 0$ 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)

1.1 BASES: FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure $\geq$ 800 psia and Core Flow $\geq$ 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 2 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.

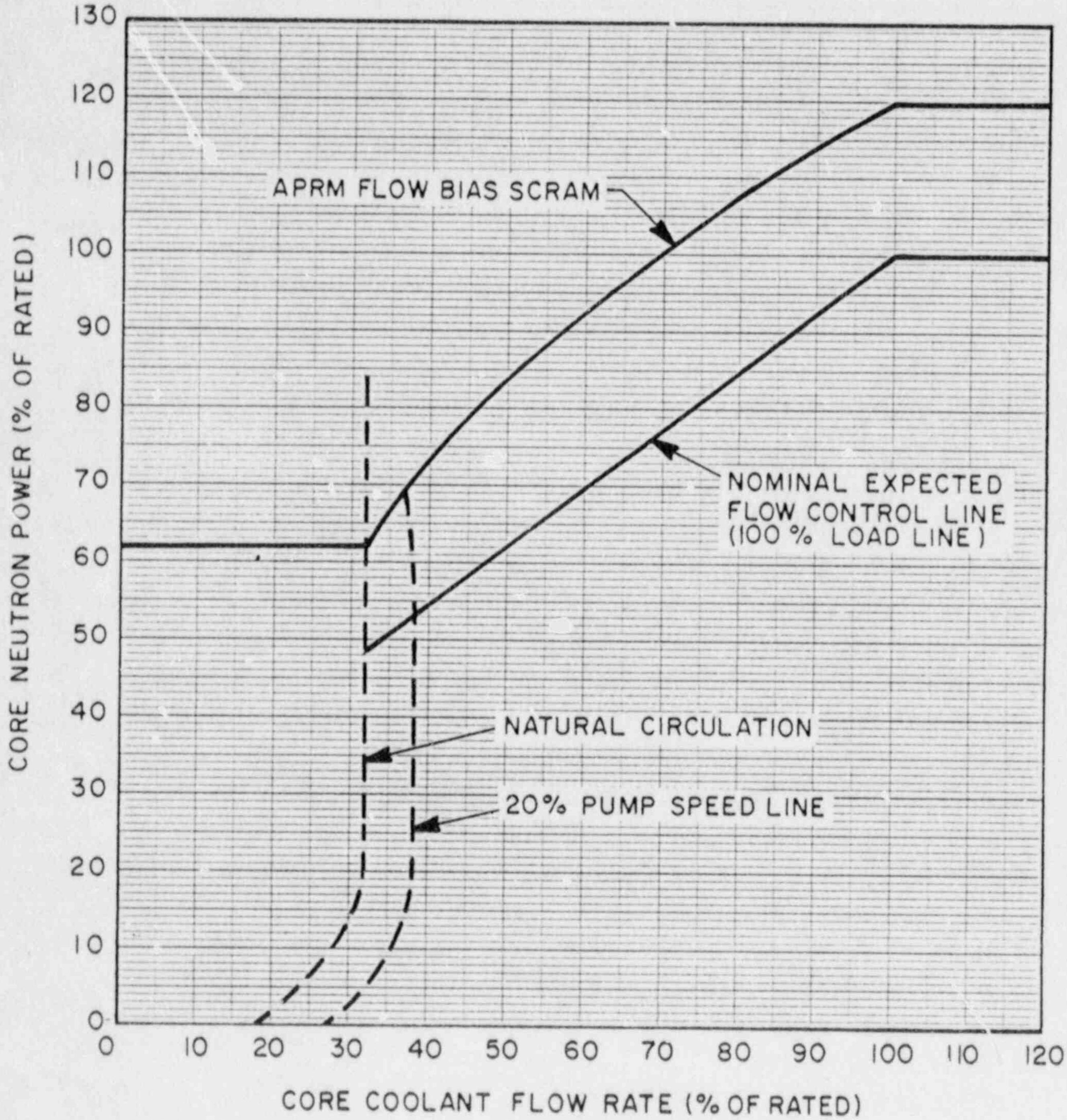
E. References

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 < 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.

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 $\leq$ LHGRd

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

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 $\geq$ 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.

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 >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)

Ni = 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:

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

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

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 (Cont'd.)

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 figure 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 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 Refil 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).

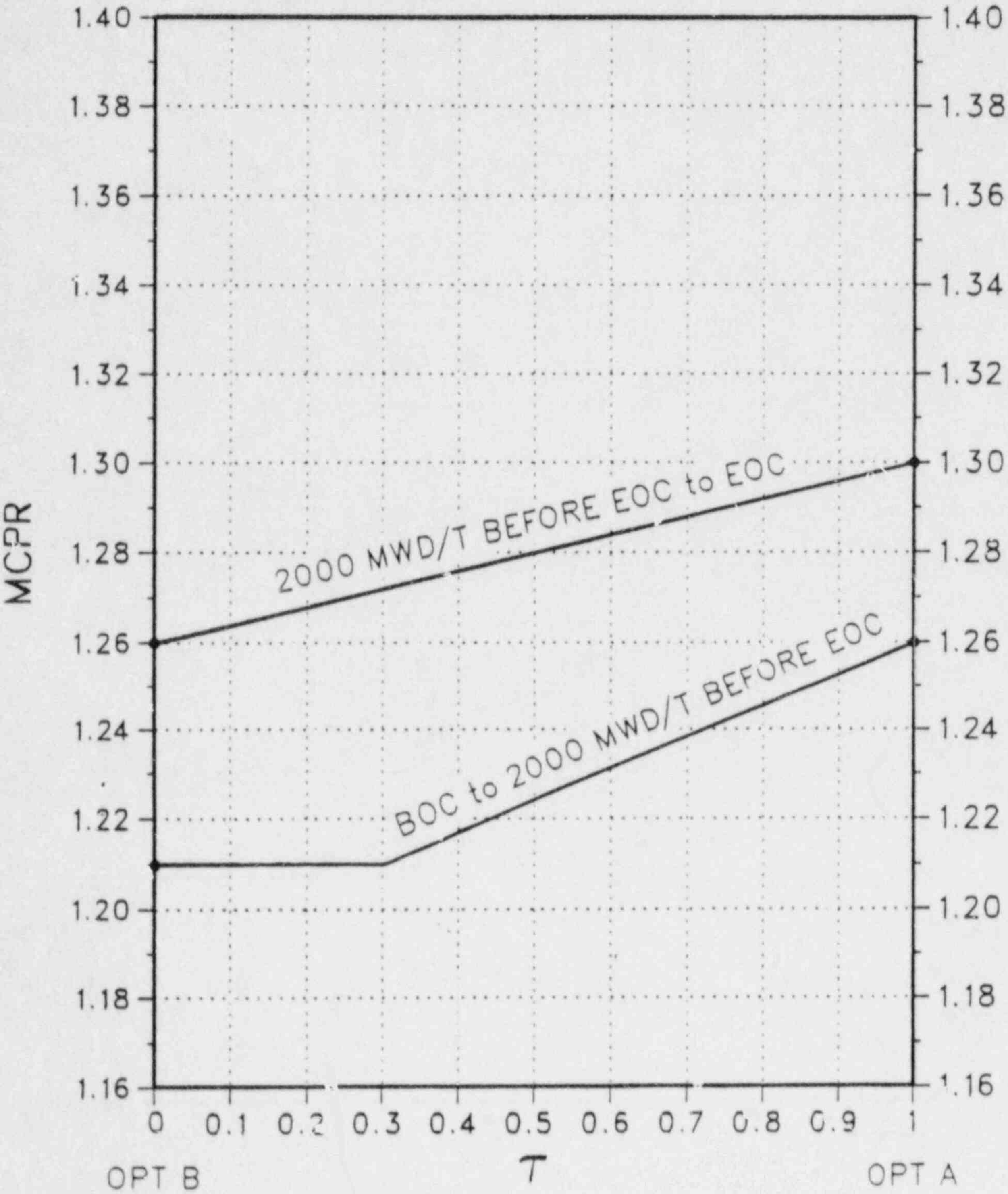
8. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, NEDO-24081, December 1977, and for Unit 3, NEDO-24082, December 1977.

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.

PEAC. ! BOTTOM UNIT 3

FIGURE 3.5.K.1

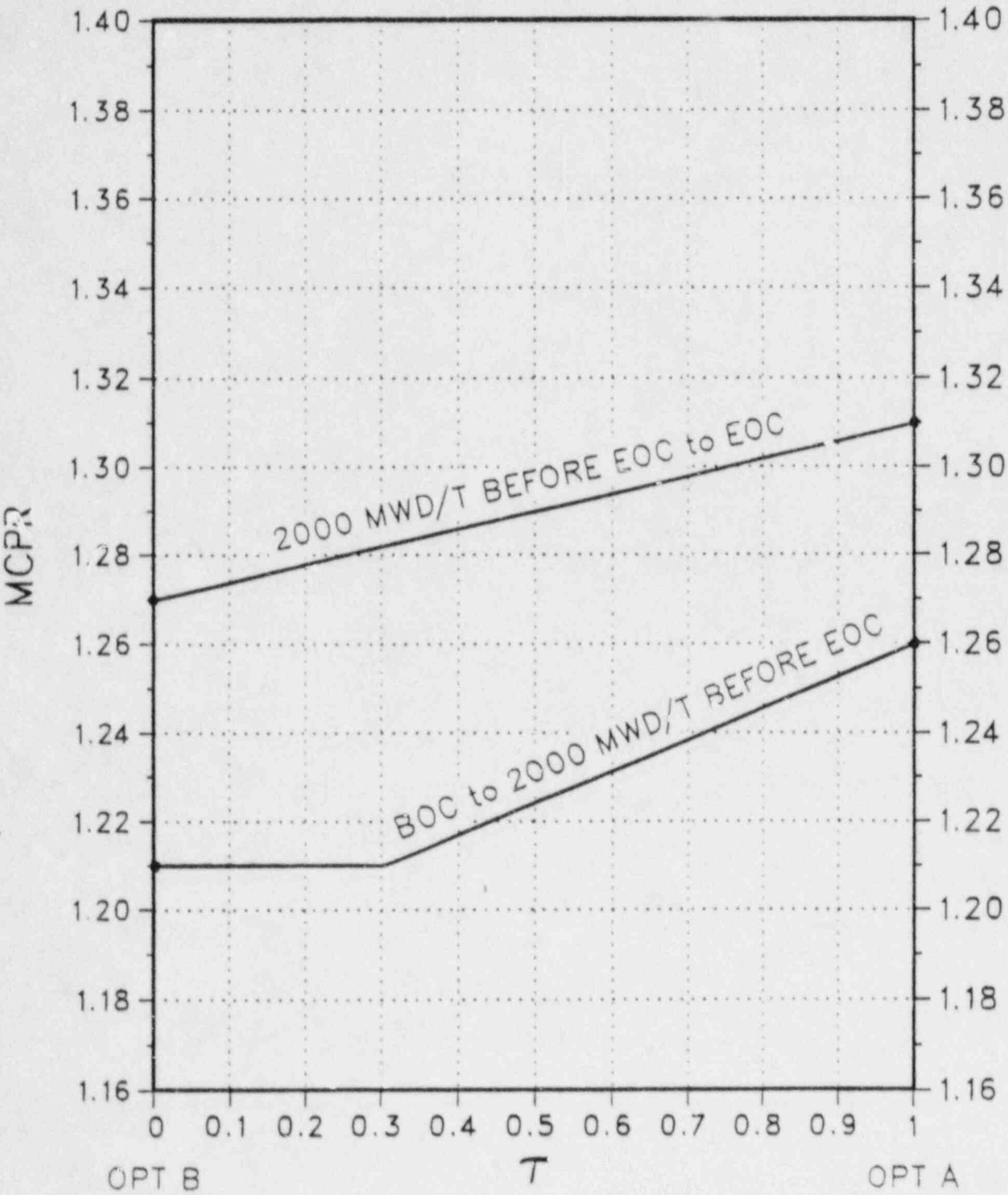
MCPR OPERATING LIMIT vs τ
FUEL TYPES: BP/P8X8R.LTA,GE8X8EB
(STANDARD OPERATING CONDITIONS)

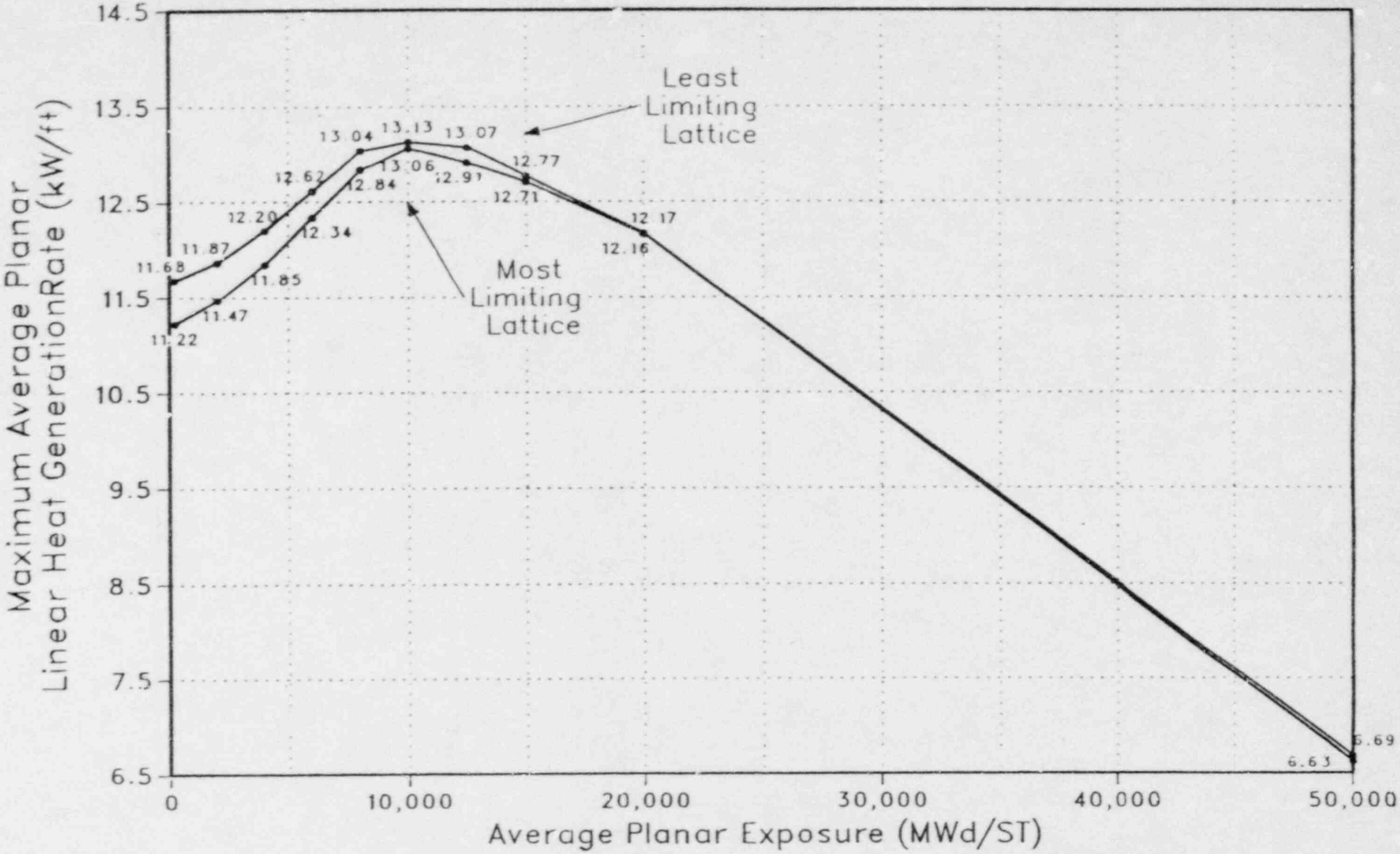


PEACH BOTTOM UNIT 3

FIGURE 3.5.K.2

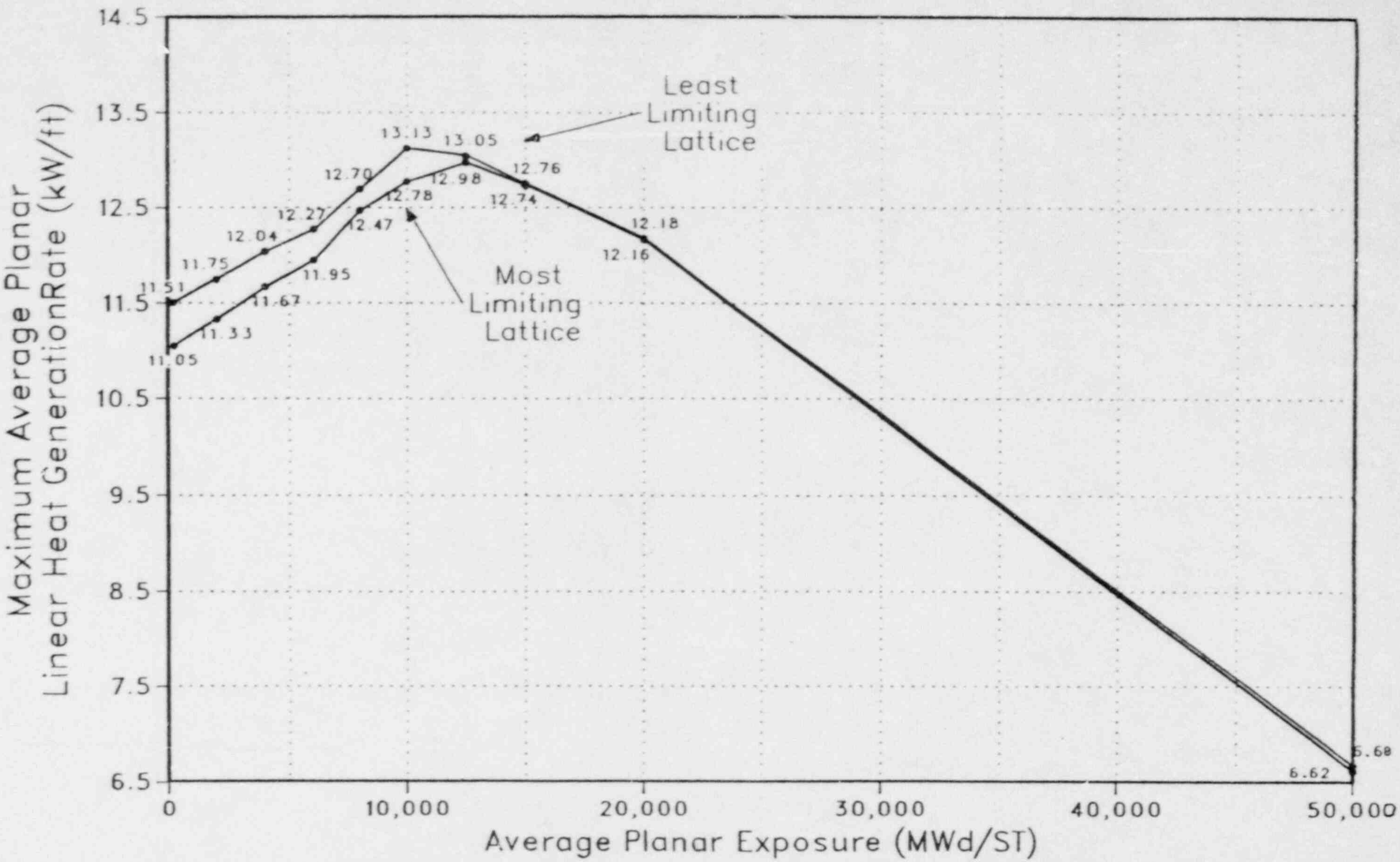
MCPR OPERATING LIMIT vs τ
FUEL TYPES: BP/P8X8R,LTA,GE8X8EB
(INCREASED CORE FLOW)





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

FIGURE 3.5.1.F



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

FIGURE 3.5.1.G

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Application were served on the following by deposit in the United States Mail, first class postage prepaid, on the 7th day of July, 1988.

William T. Russell, Regional Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

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

Mr. Thomas Gerusky, Director
Bureau of Radiological Protection
Department of Environmental Resources
P.O. Box 2063
Harrisburg, PA 17120

Eugene J. Bradley

Attorney for
Philadelphia Electric Company