ATTACHMENT 2

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 and 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

REVISED TECHNICAL SPECIFICATION PAGES

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1.0 DEFINITIONS

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

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

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

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

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

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

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

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

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

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

2.1.A (Cont'd)

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

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

where,

FRP = fraction of rated thermal power (3293 MWt)

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

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

2. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.

3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

SAFETY LIMIT

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

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).

LIMITING SAFETY SYSTEM SETTING

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 the value specified in the CORE OPERATING LIMITS REPORT.

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

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

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 References 2 and 3.

2.1 BASES: (Cont'd)

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO 10802, February 1973.

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

3. PECo-FMS-0006, "Methods for Performing BWR Reload Safety Evaluations" (latest approved revision)

NOTES FOR TABLE 3.1.1 (Cont'd)

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

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

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

    FRP = fraction of rated thermal power (3293 MWt).
    MFLPD = maximum fraction of limiting power density where the limiting power density is the value specified in the CORE OPERATING LIMITS REPORT.

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

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

    ΔW = the difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.58 ΔW) is accomplished by correcting the flow input of the flow biased High Flux trip setting to preserve the original (two loop) relationship between APRM High Flux setpoint and recirculation drive flow or by adjusting the APRM Flux trip setting. ΔW = 0 for two loop operation.

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

13. See Section 2.1.A.1.

Unit 2

TABLE 3.2.C

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
4 (2)	APRM Upscale (Flow Biased)	$\leq(0.58W+50-0.58\Delta W) \times \frac{FRP}{MFLPD}$	6 Inst. Channels	(10)
4	APRM Upscale (Startup Mode)	$\leq$12%	6 Inst. Channels	(10)
4	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(10)
1 (2)(7)(11)	Rod Block Monitor (Flow Biased)	$\leq(0.66W+(N-66)-0.66\Delta W) \times \frac{FRP}{MFLPD}$ with a maximum of $\leq$N%	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\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 the value specified in the CORE OPERATING LIMITS REPORT.

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

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

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

   ΔW is the difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between the rod block setpoint and recirculation drive flow, or by adjusting the rod block setting. $\Delta W = 0$ for two loop operation.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 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.

NOTES FOR TABLE 3.2.C (Cont.)

9. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn.

10. For the Startup (for IRM rod block) and the Run (for APRM rod block) positions of the Reactor Mode Selector Switch and with the number of OPERABLE channels:

    a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

    b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

11. The value of N is specified in the CORE OPERATING LIMITS REPORT.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.I Average Planar LHGR	4.5.I Average Planar LHGR
During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limit for the most limiting lattice (excluding natural uranium) specified in the CORE OPERATING LIMITS REPORT during two recirculation loop operations. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the reduction factors specified in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore ALPHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	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.
3.5.J Local LHGR	4.5.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.	The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

LHGR $\leq$ LHGRd

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

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

SURVEILLANCE 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 shall be as specified in the CORE OPERATING LIMITS REPORT for when

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

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

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

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

SURVEILLANCE 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 insertion time to the 20% position = 0.694 sec.

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

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

PBAPS

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

3.5 BASES (Cont'd.)

H. Engineered Safeguards Compartments Cooling and Ventilation

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

I. Average Planar LHGR

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

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

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

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

3.5. BASES (Cont'd)

J. Local LHGR

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

K. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and References 7 and 10. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not violated during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The transients evaluated are as described in References 7 and 10.

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 References 7 and 10. Input data and operating conditions used in this analysis are shown in References 7 and 10 and in the Supplemental Reload Licensing Analysis.

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

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

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

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

3.5.L. BASES (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limits value predominantely 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.

10. PECo-FMS-0006, "Methods for Performing BWR Reload Safety Evaluations" (latest approved revision)

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

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

4.5.L MCPR Limits for Core Flows Other than Rated

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

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

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

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

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

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

PBAPS

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

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

PBAPS

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

PBAPS

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

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

6.9.1 Routine Reports (Cont'd)

c. Annual Safety/Relief Valve Report

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

d. Monthly Operating Report

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

e. Core Operating Limits Report

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

a. The APLHGR for Specification 3.5.I,

b. The MCPR for Specification 3.5.K,

c. The K_f core flow adjustment factor for Specification 3.5.K,

d. The LHGR for Specification 3.5.J,

e. The upscale flow biased Rod Block Monitor setpoint and the upscale high flow clamped Rod Block monitor setpoint of Specification 3.2.C.

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

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel"

b. Philadelphia Electric Company Methodologies as described in:

(1) PECo-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code"

(2) PECo-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECo)"

6.9.1 Routine Reports (Cont'd)

(3) PECo-FMS-0003-A, "Steady-State Fuel Performance Methods Report"

(4) PECo-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis"

(5) PECo-FMS-0005, "Methods for Performing BWR Steady-State Reactor Physics Analysis"

(6) PECo-FMS-0006, "Methods for Performing BWR Reload Safety Evaluations"

(3) The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(4) The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be submitted upon issuance for each Operating Cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

LIST OF FIGURES

PBAPS

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1.0 DEFINITIONS

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

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

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

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

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

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

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

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

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

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

2.1.A (Cont'd)

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

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

where,

FRP = fraction of rated thermal power (3293 MWt)

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

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

2. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15 percent of rated power.

3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia)	B. APRM Rod Block Trip Setting SRB ≤ (0.58 W + 50% - 0.58ΔW) $\frac{(FRP)}{MFLPD}$ where: FRP = fraction of rated thermal power (3293 MWt). MFLPD = maximum fraction of limiting power density where the limiting power density is the value specified in the CORE OPERATING LIMITS REPORT. The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.
C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than minus 160 inches indicated level (378 inches above vessel zero).	C. Scram and isolation--reactor low water level ≥ 538 in. above vessel zero (0" on level instruments)

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 References 4 and 5.

2.1 BASES (Cont'd)

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO 10802, February 1973.

2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", NEDO 24154 and NEDE 24154-P, Volumes I, II, and III.

3. "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors NEDO-24154 and NEDE 24154-P, Volumes I, II, and III.

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

5. PECo-FMS-0006, "Methods for Performing BWR Reload Safety Evaluations" (latest approved revision)

NOTES FOR TABLE 3.1.1 (Cont'd)

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

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

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

    FRP = fraction of rated thermal power (3293 MWt).
    MFLPD = maximum fraction of limiting power density where the limiting power density is the value specified in the CORE OPERATING LIMITS REPORT.

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

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

    Delta W = The difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.58 delta W) is accomplished by correcting the flow input of the flow biased High Flux trip setting to preserve the original (two loop) relationship between APRM High Flux setpoint and recirculation drive flow or by adjusting the APRM Flux trip setting. Delta W equals zero for two loop operation.

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

13. See Section 2.1.A.1.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
4 (2)	APRM Upscale (Flow Biased)	$\leq(0.58W+50-0.58\Delta W) \times \frac{FRP}{MFLPD}$	6 Inst. Channels	(10)
4	APRM Upscale (Startup Mode)	$\leq$12%	6 Inst. Channels	(10)
4	APRM Downscale	$\geq$2.5 indicated on scale	6 Inst. Channels	(10)
1 (2)(7)(11)	Rod Block Monitor (Flow Biased)	$\leq(0.66W+(N-66)-0.66\Delta W) \times \frac{FRP}{MFLPD}$ with a maximum of $\leq$N%	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\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 the value specified in the CORE OPERATING LIMITS REPORT.

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

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

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

   ΔW is the difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between the rod block setpoint and recirculation drive flow, or by adjusting the rod block setting. $\Delta W = 0$ for two loop operation.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 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.

NOTES FOR TABLE 3.2.C (Cont.)

9. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour. This note is applicable in the "Run" mode, the "Startup" mode and the "Refuel" mode if more than one control rod is withdrawn.

10. For the Startup (for IRM rod block) and the Run (for APRM rod block) positions of the Reactor Mode Selector Switch and with the number of OPERABLE channels:

    a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

    b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

11. The value of N is specified in the CORE OPERATING LIMITS REPORT.

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

3.5.J Local LHGR

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

LHGR $\leq$ LHGRd

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

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

SURVEILLANCE 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 shall be as specified in the CORE OPERATING LIMITS REPORT for when

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

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

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

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

SURVEILLANCE 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

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

3.5 BASES (Continued)

H. Engineered Safeguards Compartments Cooling and Ventilation

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

I. Average Planar LHGR

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

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

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

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

3.5 BASES (Cont'd)

J. Local LHGR

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

K. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and References 7 and 10. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not violated during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The transients evaluated are as described in References 7 and 10.

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 References 7 and 10. Input data and operating conditions used in this analysis are shown in References 7 and 10 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 indication 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 experi nce has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominantely 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.

10. PECo-FMS-0006, "Methods for Performing BWR Reload Safety Evaluations" (latest approved revision)

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

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

4.5.L MCPR Limits for Core Flows Other than Rated

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

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

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

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

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

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

PBAPS

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

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

6.9.1 Routine Reports (Cont'd)

c. Annual Safety/Relief Valve Report

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

d. Monthly Operating Report

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

e. Core Operating Limits Report

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

a. The APLHGR for Specification 3.5.I,

b. The MCPR for Specification 3.5.K,

c. The K_f core flow adjustment factor for Specification 3.5.K,

d. The LHGR for Specification 3.5.J,

e. The upscale flow biased Rod Block Monitor setpoint and the upscale high flow clamped Rod Block monitor setpoint of Specification 3.2.C.

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

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel"

b. Philadelphia Electric Company Methodologies as described in:

(1) PECo-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code"

(2) PECo-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECo)"

6.9.1 Routine Reports (Cont'd)

(3) PECo-FMS-0003-A, "Steady-State Fuel Performance Methods Report"

(4) PECo-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis"

(5) PECo-FMS-0005, "Methods for Performing BWR Steady-State Reactor Physics Analysis"

(6) PECo-FMS-0006, "Methods for Performing BWR Reload Safety Evaluations"

(3) The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(4) The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be submitted upon issuance for each Operating Cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.