

PBAPS

NOTES FOR TABLE 3.2.C

1. For the start~~up~~ and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.
2. The equation for Trip Level Setting will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP) where:

FRP = fraction of rated thermal power (3293 MWt)

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

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

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

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

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

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

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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 Reference 7. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

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

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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.

PBAPS6.9.1 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 Kf core flow adjustment factor for Specification 3.5.K,
- d. The LHGR for Specification 3.5.J,
- e. The upscale flow biased Rod Block Monitor setpoint and the upscale high flow clamped Rod Block monitor setpoint of Specification 3.2.C.

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

- a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version)
- b. Philadelphia Electric Company Methodologies as described in:
 - (1) PECO-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using FIBWR Computer Code"
 - (2) PECO-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECo)"

PBAPS6.9.1 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-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"
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- (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.

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