SUSQUEHANNA SES UNIT 2 CYCLE 5

TECHNICAL SPECIFICATION CHANGES

SEPTEMBER 1990

PENNSYLVANIA POWER & LIGHT COMPANY

PP&L

SECTION

LIST OF TABLES

LIST OF FIGURES

RATED THERMAL POWER

1.33 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until deenergization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.35 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.36 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

Insert 1 →

SECONDARY CONTAINMENT INTEGRITY

1.37 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows, resilient material seals, or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1a.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 ~~The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06*~~ W ~~w~~ith the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow. Insert 2

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

Insert 3

With MCPR less than ~~1.06*~~ and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

~~*See Specification 3.4.1.1.2.a for single loop operation requirement.~~

2.1 SAFETY LIMITS

BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are ~~the~~ principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. ~~Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specification 2.1.2 for ANF fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.~~ The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. ~~Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0.~~ These ~~conditions~~ stresses represent a significant departure from the condition intended by design for planned operation. ~~The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling (ref. XN-NF-524(A) Revision 1).~~

Insert 4

Insert 5

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the XN-3 correlation is valid for critical power calculations at pressures greater than 580 psig and bundle mass fluxes greater than 0.25×10^6 lbs/hr-ft^2. For operation at low pressures or low flows, the fuel cladding integrity Safety Limit is established by a limiting condition on core THERMAL POWER with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to assure a minimum bundle flow for all fuel assemblies which have a relatively high power and potentially can approach a critical heat flux condition. For the ANF 9 x 9 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For this design, the coolant minimum flow and maximum flow area is such that the mass flux is always greater than 0.25×10^6 lbs/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lbs/hr-ft^2 is 3.35 Mwt or greater. At 25% thermal power a bundle power of 3.35 Mwt corresponds to a bundle radial peaking factor of greater than 3.0 which is significantly higher than the expected peaking factor Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressures below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR).

Insert 6 → ~~The Safety Limit MCPR~~ assures sufficient conservatism in the operating ~~MCPR limit that~~ in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. ~~The margin between calculated boiling transition (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed~~ The statistical procedure ~~which~~ considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit Type analyses is the uncertainty inherent in the XN-3 critical power correlation. XN-NF-524 (A), Revision 1 describes the methodology used in determining the ~~Safety Limit MCPR~~ Insert 7 discussed in PL-NF-90-001

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the XN-3 correlation (refer to Section B 2.1.1), the assumed reactor conditions used in ~~defining~~ Calculation the safety limit introduce conservatism into the ~~limit~~ because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that ~~during sustained operation at the Safety Limit MCPR there would be no transition boiling in the core.~~ Insert 8 If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicates that LWR fuel can survive for an extended period of time in an environment of boiling transition.

evaluating the approach to

Insert 9 → Here

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the greater of the two values determined from Figure 3.2.3-1 and Figure 3.2.3-2.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With MCPR less than the applicable MCPR limit determined above, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit determined from Figure 3.2.3-1 and Figure 3.2.3-2:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

REPLACE WITH THE FOLLOWING THREE PAGES.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the greater of:

a) The Flow-Dependent MCPR value determined from Figure 3.2.3-1, and

b) The Power-Dependent MCPR value determined from the following equation:

$$MCPR = MCPR_B + (MCPR_A - MCPR_B) \times \text{SCRAM SPEED FRACTION}$$

where:

$MCPR_A$ and $MCPR_B$ are determined from curve A and curve B of one of the following figures, as appropriate:

Figure 3.2.3-2: EOC-RPT and Main Turbine Bypass Operable

Figure 3.2.3-3: Main Turbine Bypass Inoperable

Figure 3.2.3-4: EOC-RPT Inoperable

SCRAM SPEED FRACTION is a number between 0.0 and 1.0 (inclusive) based on measured core average scram speed. This fraction is used to interpolate between Curve B MCPR values corresponding to an average scram speed of 4.2 feet/second and Curve A MCPR values corresponding to the maximum allowed core average scram insertion times given in Specification 3.1.3.3. The SCRAM SPEED FRACTION is obtained from Table 3.2.3-1.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With MCPR less than the applicable MCPR limit determined above, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.2.3 MCPR, with

a) SCRAM SPEED FRACTION = 1.0 prior to the performance of the initial scram time measurement for the cycle in accordance with Specification 4.1.3.2(a), or

b) SCRAM SPEED FRACTION as determined from Table 3.2.3-1 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2

shall be determined to be greater than or equal to the applicable MCPR limit determined from Figure 3.2.3-1 and the applicable figure selected from Figures 3.2.3-2 through 3.2.3-4:

1) At least once per 24 hours,

2) Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

3) Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

4) The provisions of Specification 4.0.4 are not applicable.

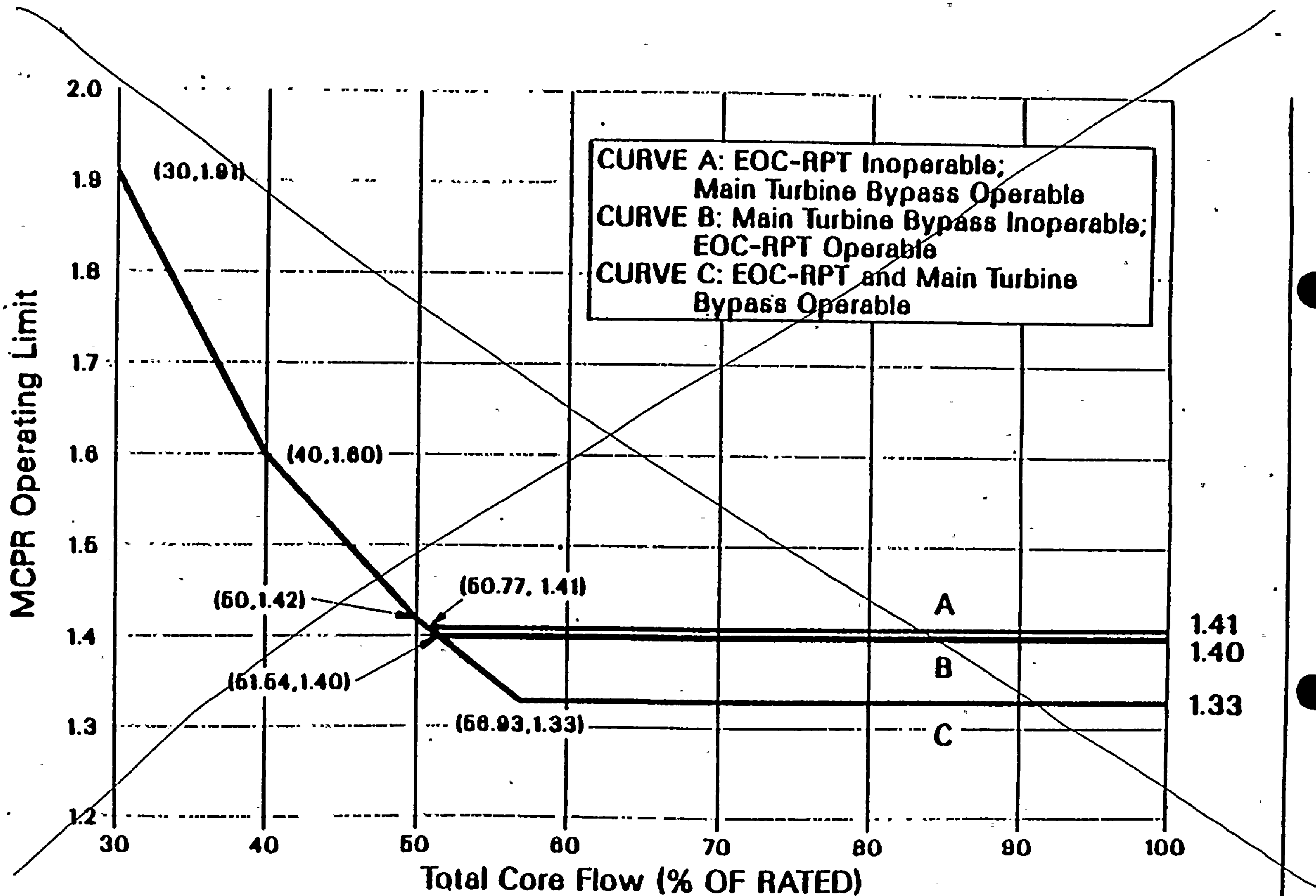
TABLE 3.2.3-1

SCRAM SPEED FRACTION
VERSUS AVERAGE SCRAM TIMES

Rod Positions	MAXIMUM TIMES TO ROD POSITIONS (Seconds)					
45	.38	.39	.40	.41	.42	.43
39	.74	.76	.79	.82	.85	.86
25	1.57	1.63	1.70	1.78	1.87	1.93
5	2.76	2.88	3.01	3.16	3.32	3.49
SCRAM SPEED FRACTION	0.0	0.2	0.4	0.6	0.8	1.0

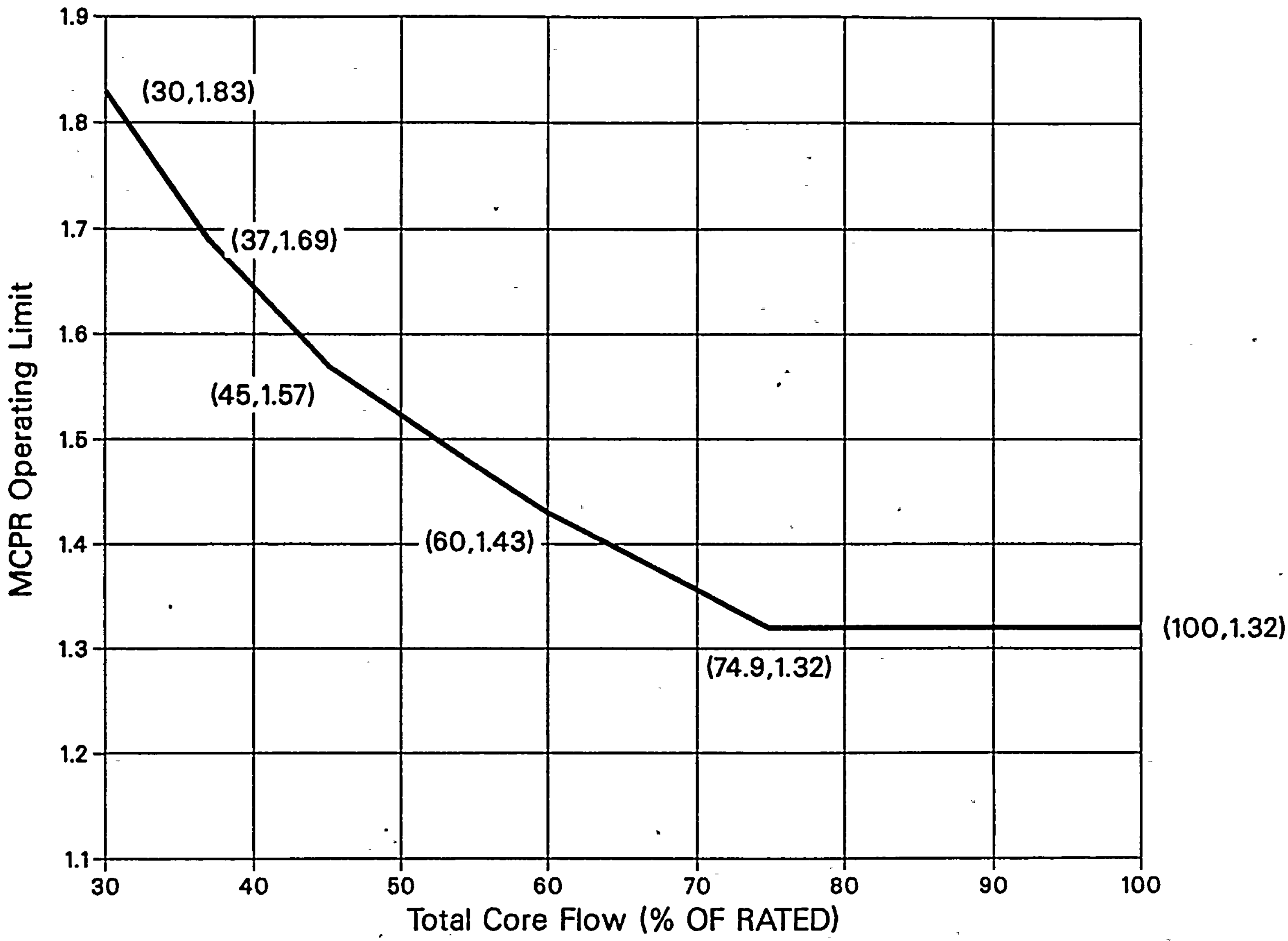
NOTE: Determine SCRAM SPEED FRACTION from farthest left-hand column whose listed values are all greater than measured average scram times using most recent measurement for each rod.

REPLACE WITH THE FOLLOWING PAGE



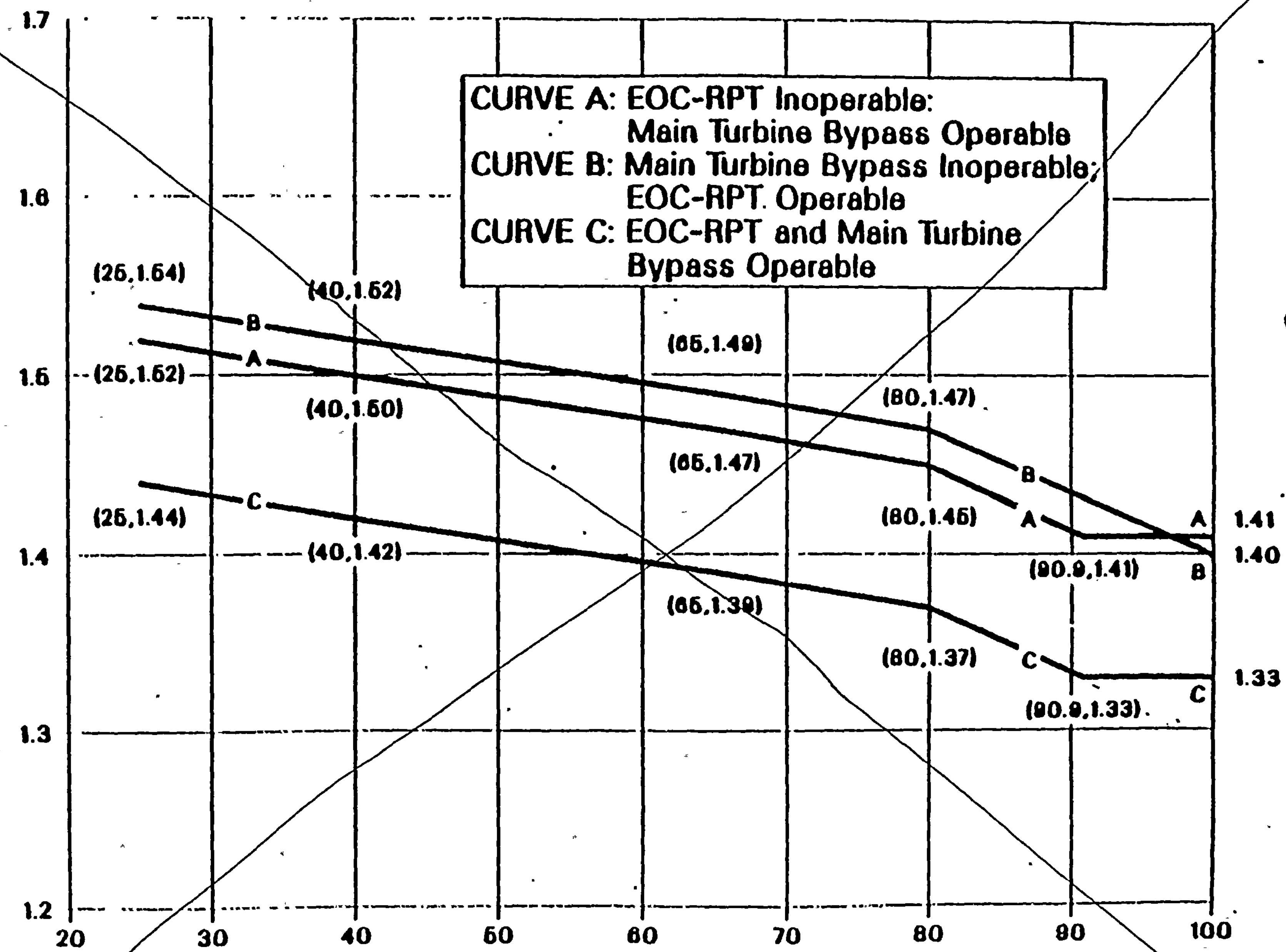
FLOW DEPENDENT MCPR OPERATING LIMIT

FIGURE 3.2.3-1

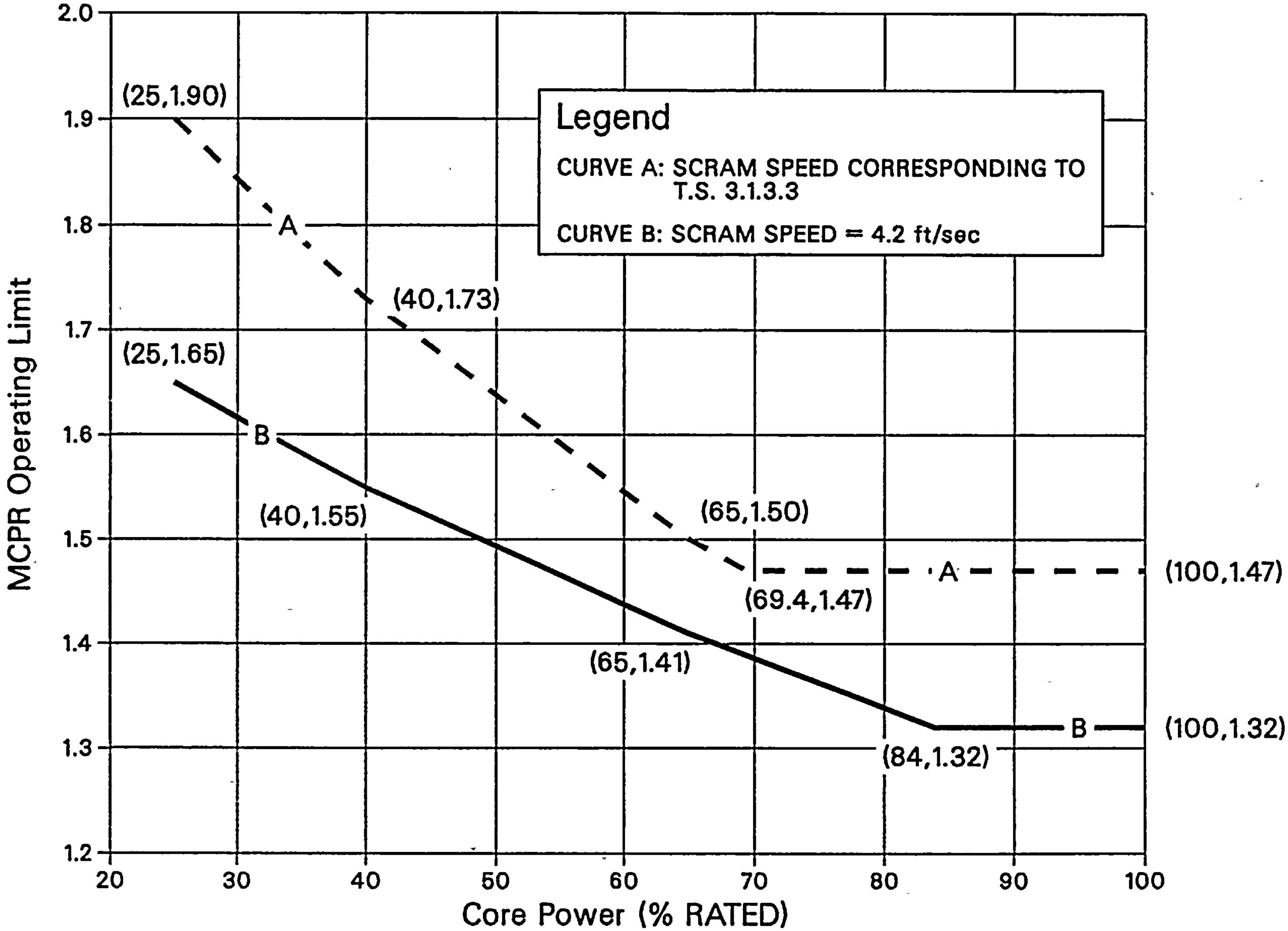


FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1

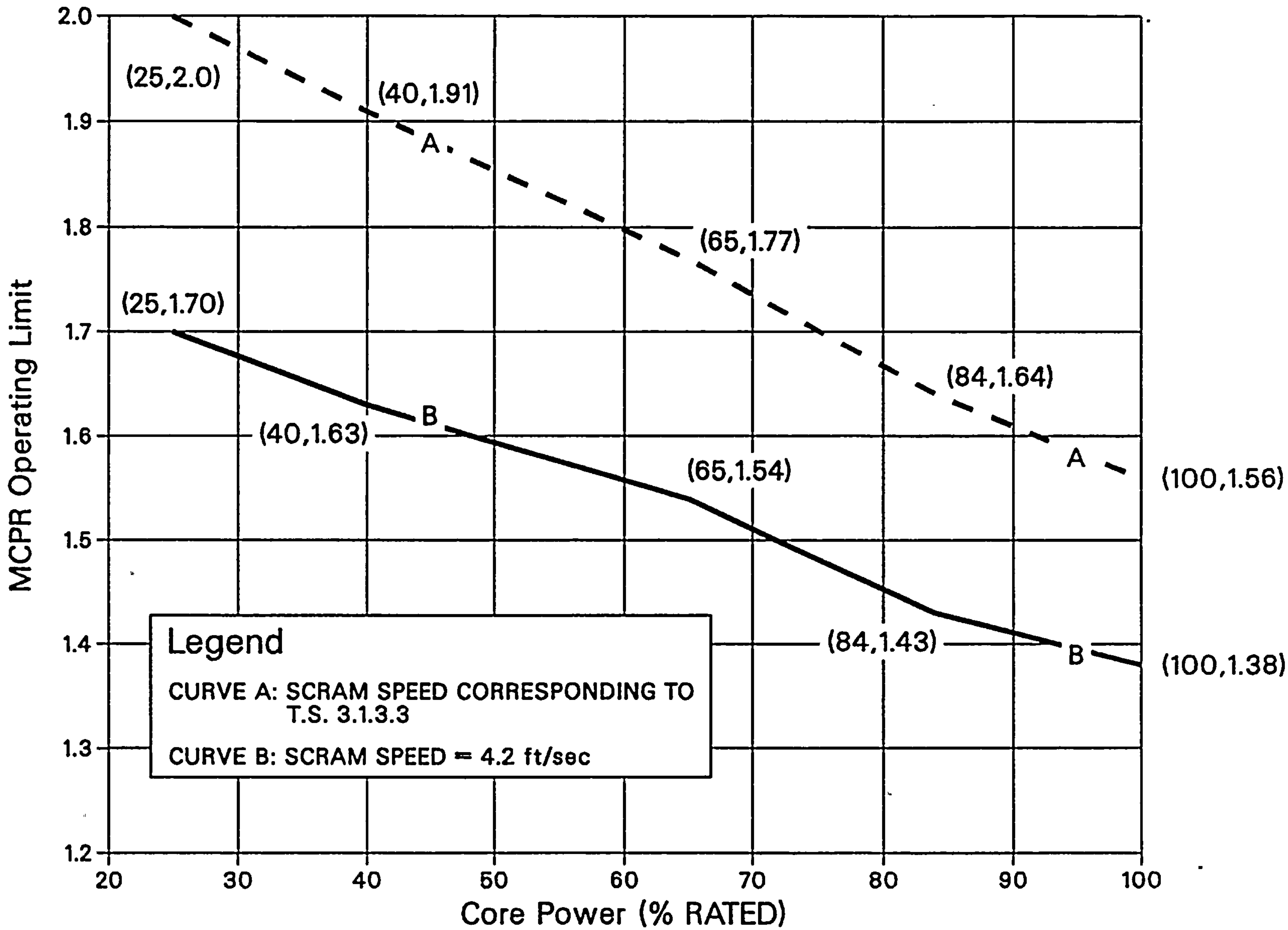
REPLACE WITH THE FOLLOWING THREE PAGES



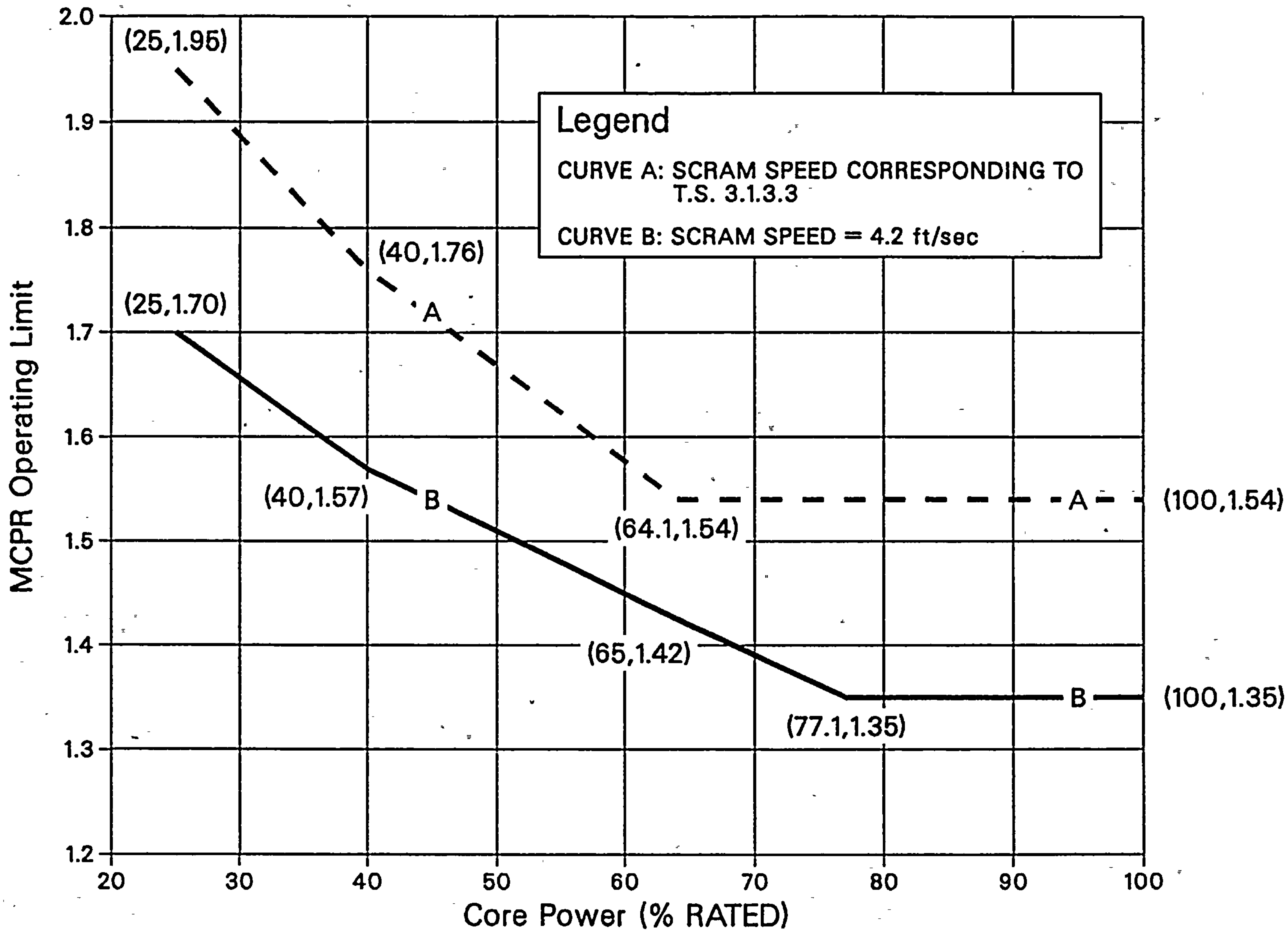
REDUCED POWER MCPR OPERATING LIMIT
FIGURE 3.2.3-2



REDUCED POWER MCPR OPERATING LIMIT
EOC-RPT AND MAIN TURBINE BYPASS OPERABLE
FIGURE 3.2.3-2



REDUCED POWER MCPR OPERATING LIMIT
MAIN TURBINE BYPASS INOPERABLE
FIGURE 3.2.3-3



REDUCED POWER MCPR OPERATING LIMIT
EOC-RPT INOPERABLE
FIGURE 3.2.3-4

Figure 3.4.1.1.1-1
THERMAL POWER RESTRICTIONS

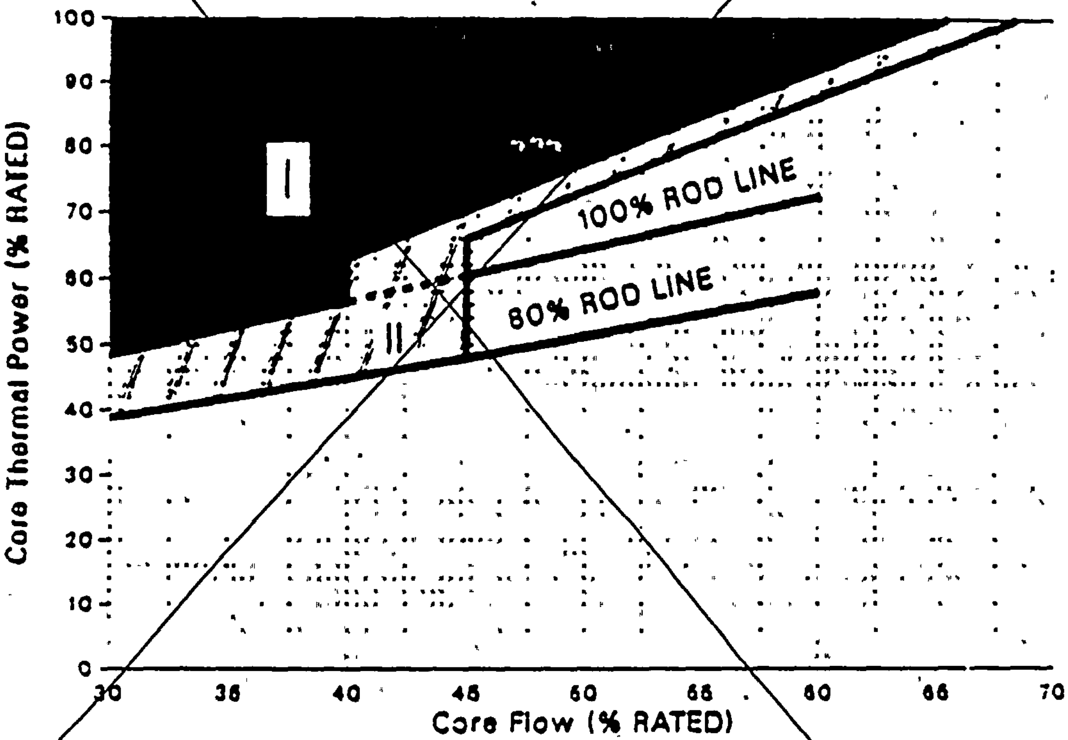


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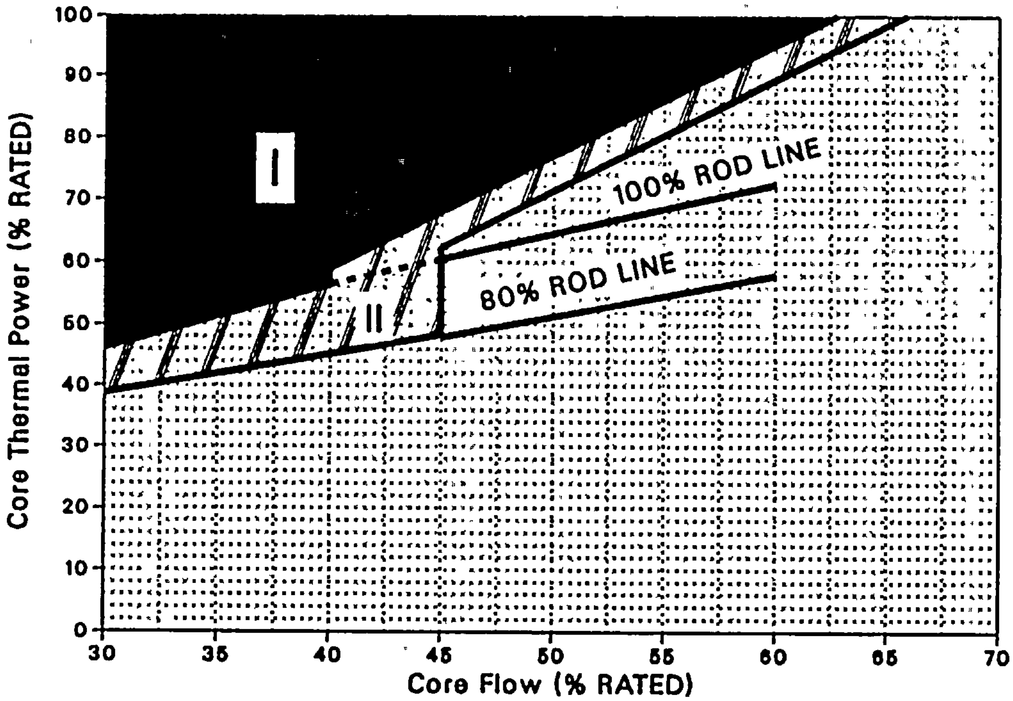
Figure 3.4.1.1.1-1
THERMAL POWER RESTRICTIONS



REACTOR COOLANT SYSTEM

RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed ≤ 80% of the rated pump speed and the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1, and

a. the following revised specification limits shall be followed:

~~1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.~~

1 ~~2~~. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
≤ 0.58W + 54%	≤ 0.58W + 57%.

2 ~~3~~. Specification 3.2.2: the APRM Setpoints shall be as follows:

Trip Setpoint	Allowable Value
$S \le (0.58W + 54\%)T$	$S \le (0.58W + 57\%)T$
$S_{RB} \le (0.58W + 45\%)T$	$S_{RB} \le (0.58W + 48\%)T$

3 ~~4~~. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the largest of the following values:

a. the MCPR determined from Figure 3.2.3-1 plus 0.01, and

b. the MCPR determined from Figure 3.2.3-2 OR FIGURE 3.2.3-3 OR FIGURE 3.2.3-4, AS APPROPRIATE plus 0.01.

4 ~~5~~. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

		Trip Setpoint	Allowable Value
a.	RBM - Upscale	≤ 0.66W + 36%	≤ 0.66W + 39%
b.	APRM-Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*+, except during two loop operation.#

ACTION:

a. In OPERATIONAL CONDITION 1:

1. With

a) no reactor coolant system recirculation loops in operation, or

b) Region I of Figure 3.4.1.1.1-1 entered, or

c) Region II of Figure 3.4.1.1.1-1 entered and core thermal hydraulic instability occurring as evidenced by:

LIMITING CONDITION FOR OPERATION (Continued)

f. With any pump discharge bypass valve not OPERABLE close the valve and verify closed at least once per 31 days.

SURVEILLANCE REQUIREMENTS

4.4.1.1.2.1 Upon entering single loop operation and at least once per 24 hours thereafter, verify that the pump speed in the operating loop is ≤ 80% of the rated pump speed.

4.4.1.1.2.2 At least 50% of the required LPRM upscale alarms shall be determined OPERABLE by performance of the following on each LPRM upscale alarm.

1) CHANNEL FUNCTIONAL TEST at least once per 92 days, and

2) CHANNEL CALIBRATION at least once per 184 days.

4.4.1.1.2.3 Within 15 minutes prior to either THERMAL POWER increase resulting from a control rod withdrawal or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30%**** of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50%**** of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b.## ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c.## ≤ 50°F between the reactor coolant within the loop not in operation and operating loop.

4.4.1.1.2.4 The pump discharge valve and bypass valve in both loops shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2.5 The pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 102.5% and 105%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.2.6 During single recirculation INOPERABLE loop operation, all jet pumps, including those in the ~~operable~~ loop, shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur:###

a. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established single recirculation pump speed-loop flow characteristics.

SURVEILLANCE REQUIREMENTS (Continued)

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop patterns by more than 10%.

4.4.1.1.2.7 The SURVEILLANCE REQUIREMENTS associated with the specifications referenced in 3.4.1.1.2a shall be followed.

* See Special Test Exception 3.10.4.

** If not performed within the previous 31 days.

**** Initial value. Final value to be determined based on startup testing. Any required change to this value shall be submitted to the Commission within 90 days of test completion.

\# See Specification 3.4.1.1.1 for two loop operation requirements.

\## This requirement does not apply when the loop not in operation is isolated from the reactor pressure vessel.

\### During startup testing following each refueling outage, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the performance of subsequent required surveillances.

\+ The LPRM upscale alarms are not required to be OPERABLE to meet this specification in OPERATIONAL CONDITION 2.

b. The indicated total core flow differs by more than 10% from the established total core flow value from single recirculation loop flow measurements.

REACTIVITY CONTROL SYSTEMS

BASES

REACTIVITY ANOMALIES (Continued)

Since the comparisons are easily done, frequent checks are not an imposition on normal operation. A 1% deviation in reactivity from that of the predicted is larger than expected for normal operation, and therefore should be thoroughly evaluated. A deviation as large as 1% would not exceed the design conditions of the reactor.

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to ~~prevent the MCPR from becoming less than the limit specified in Specification 2.1.2 during the core wide transient analyzed in the cycle specific transient analysis report. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the limit specified in Specification 2.1.2.~~ Insert 10 The occurrence of scram times longer then those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a

CONTROL ROD PROGRAM CONTROLS (Continued)

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

Parametric Control Rod Drop Accident analyses have shown that for a wide range of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient, effective delayed neutron fraction, and maximum four-bundle local peaking factor are compared with the inputs to the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the 280 cal/gm design limit to demonstrate compliance for each operating cycle. If cycle-specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle-specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in PL-NF-90-001 AND XN-NF-80-19 Volume 1.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum quantity of 4587 gallons of sodium pentaborate solution containing a minimum of 5500 lbs. of sodium pentaborate is required to meet this shutdown requirement. There is an additional allowance of 165 ppm in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel. The temperature requirement for the sodium penetrate solution is necessary to ensure that the sodium penetaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

~~The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.~~

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46.

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 dependent only secondarily on the rod to rod power distribution within an assembly. The Technical Specification APLHGR for ANF fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is shown in Figure 3.2.1-1.

The calculational procedure used to establish the APLHGR shown on Figure 3.2.1-1 is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. These models are described in XN-NF-80-19, Volumes 2, 2A, 2B and 2C.

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analyses. In addition, the APRM setpoints must be adjusted to ensure that ≥1% plastic strain and fuel centerline melting do not occur during the worst anticipated operational occurrence (AOO), including transients initiated from partial power operation.

For ANF fuel the T factor used to adjust the APRM setpoints is based on the FLPD calculated by dividing the actual LHGR by the LHGR obtained from Figure 3.2.2-1. The LHGR versus exposure curve in Figure 3.2.2-1 is based on ANF's Protection Against Fuel Failure (PAFF) line shown in Figure 3.4 of XN-NF-85-67(A), Revision 1. Figure 3.2.2-1 corresponds to the ratio of PAFF/1.2 under which cladding and fuel integrity is protected during AOO's.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from ~~the established fuel cladding integrity Safety Limit MCPR, and an~~ analyses of abnormal operational transients. For any abnormal ~~operating~~ operational 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~~ Insert 11 assuming instrument trip settings given in Specification 2.2.

Insert 12 ~~To assure that the fuel cladding integrity Safety Limit is not exceeded 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 type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. ~~The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figures 3.2.3-1 and 3.2.3-2.~~ Insert 13

Insert 14 ~~The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to an ANF core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and non-pressurization events are described in XN-NF-79-71 and XN-NF-84-105. The principal result of this evaluation is the reduction in MCPR caused by the transient.~~

Figure 3.2.3-1 defines core flow dependent MCPR operating limits which assure that ~~the Safety Limit MCPR will not be violated~~ Insert 15 during a flow increase transient resulting from a motor-generator speed control failure. The flow dependent MCPR is only calculated for the manual flow control mode. Therefore, automatic flow control operation is not permitted. ~~Figure 3.2.3-2 defines the power dependent MCPR operating limit which assures that the Safety limit MCPR will not be violated in the event of a Feedwater Controller Failure, Rod Withdrawal Error, or Load Reject Without Main Turbine Bypass operable initiated from a reduced power condition.~~ Insert 16

Cycle specific analyses are performed for the most limiting local and core wide transients to determine thermal margin. Additional analyses are performed to determine the MCPR operating limit with either the Main Turbine Bypass inoperable or the EOC-RPT inoperable. Analyses to determine thermal margin with both the EOC-RPT inoperable and Main Turbine Bypass inoperable have not been performed. Therefore, operation in this condition is not permitted.

MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, 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 indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation ~~will be~~ was made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin ~~will thus be~~ was demonstrated such that future MCPR evaluation below this power level ~~will be shown to be~~ is unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of 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 THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and found acceptable, provided that the unit is operated in accordance with Specification 3.4.1.1.2.

LOCA analyses for two loop operating conditions, which result in Peak Cladding Temperatures (PCTs) below 2200°F, bound single loop operating conditions. Single loop operation LOCA analyses using two-loop MAPLHGR limits result in lower PCTs. Therefore, the use of two-loop MAPLHGR limits during single loop operation assures that the PCT during a LOCA event remains below 2200°F.

THERMAL POWER, High Pressure High Flow

The MINIMUM CRITICAL POWER RATIO (MCPR) limits for single loop operation assure that the Safety Limit MCPR is not exceeded for any Anticipated Operational Occurrence (AOO).

Insert 17

For single loop operation, the RBM and APRM setpoints are adjusted by a 8.5% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

Surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive reactor vessel internals vibration. Surveillance on differential temperatures below the threshold limits of THERMAL POWER or recirculation loop flow mitigates undue thermal stress on vessel nozzles, recirculation pumps and the vessel bottom head during extended operation in the single loop mode. The threshold limits are those values which will sweep up the cold water from the vessel bottom head.

Specifications have been provided to prevent, detect, and mitigate core thermal hydraulic instability events. These specifications are prescribed in accordance with NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)," dated December 30, 1988. The boundaries of the regions in Figure 3.4.1.1.1-1 are determined using ANF decay ratio calculations and supported by Susquehanna SES stability testing.

LPRM upscale alarms are required to detect reactor core thermal hydraulic instability events. The criteria for determining which LPRM upscale alarms are required is based on assignment of these alarms to designated core zones. These core zones consist of the level A, B and C alarms in 4 or 5 adjacent LPRM strings. The number and location of LPRM strings in each zone assure that with 50% or more of the associated LPRM upscale alarms OPERABLE sufficient monitoring capability is available to detect core wide and regional oscillations. Operating plant instability data is used to determine the specific LPRM strings assigned to each zone. The core zones and required LPRM upscale alarms in each zone are specified in appropriate procedures.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design basis accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

5.3 REACTOR CORE

FUEL ASSEMBLIES

Insert 18

5.3.1 The reactor core shall contain 764 fuel assemblies, ~~with each fuel assembly containing or~~ 79 fuel rods and two water rods, ~~clad with Zircaloy -2.~~ Each fuel rod shall have a nominal active fuel length of 150 inches. Reload fuel shall have a maximum average enrichment of 4.0 weight percent U-235.

be clad with Zircaloy-2 and

~~CONTROL ROD ASSEMBLIES~~

Insert 19

~~5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.~~

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.

2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal T_{ave} of 528°F.

INSERTS

Insert 1

SCRAM SPEED FRACTION is a number between 0.0 and 1.0 (inclusive) based on measured core average scram speed. The SCRAM SPEED FRACTION is used to determine the Power-Dependent MCPR value in Specification 3.2.3 and can be obtained from Table 3.2.3-1.

Insert 2

, THERMAL POWER will be limited such that at least 99.9% of the fuel rods are not expected to experience boiling transition.

Insert 3

99.9% of the fuel rods expected to avoid boiling transition

Insert 4

The THERMAL POWER, High Pressure High Flow, Safety Limit is defined to be a core condition such that at least 99.9% of the fuel rods are not expected to experience boiling transition, which represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

Insert 5

The MCPR operating limits assure that, during normal operation and during anticipated operational occurrences the Safety Limit will not be violated. In other words, at least 99.9% of the fuel rods in the core would not be expected to experience boiling transition. The methodology used to establish the MCPR operating limits is described in PL-NF-90-001.

Insert 6

The MCPR operating limit assures sufficient conservatism in plant operation such that,

Insert 7

fraction of fuel rods in boiling transition for various MCPR values.

Insert 8

during an anticipated operational occurrence at least 99.9% of the fuel rods would not be expected to experience boiling transition.

Insert 9

ANF fuel is monitored using the XN-3 Critical Power Correlation. ANF has determined that this correlation provides sufficient conservatism to preclude the need for any penalty due to channel bow. The conservatism has been evaluated by ANF to be greater than the maximum expected ΔCPR (0.02) due to channel bow in C-lattice plants using channels for only one bundle lifetime. Since Susquehanna SES Unit 2 is a C-lattice plant and uses channels for only one bundle lifetime, monitoring of the MCPR limit with the XN-3 Critical Power Correlation is conservative with respect to channel bow and addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

Insert 10

assure that at least 99.9% of the fuel rods are not expected to experience boiling transition. The MCPR operating limits are adjusted based on measured scram time data in Specification 3.2.3 to assure the validity of the transient analyses.

Insert 11

degradation in thermal margin be such that at least 99.9% of the fuel rods in the core are not expected to experience boiling transition,

Insert 12

To assure that at least 99.9% of the fuel rods are not expected to experience boiling transition during any anticipated operational occurrence, the most limiting transients have been analyzed.

Insert 13

The limiting transient yields the largest required MCPR operating limit. The required MCPR operating limits as functions of core power, core flow, and plant equipment availability condition are presented in Figures 3.2.3-1 through 3.2.3-4.

Insert 14

The transient analyses to determine the MCPR operating limits are performed using methods described in PL-NF-90-001. Certain of the pressurization transients are analyzed statistically assuming a scram insertion versus time curve which is faster than the Technical Specification 3.1.3.3 limits. The MCPR operating limits are adjusted based on measured scram time data.

Insert 15

at least 99.9% of the fuel rods are not expected to experience boiling transition

Insert 16

Figures 3.2.3-2, 3.2.3-3, and 3.2.3-4 define the power dependent MCPR operating limits which assure that at least 99.9% of the fuel rods are not expected to experience boiling transition during the limiting event (i.e., Feedwater Controller Failure, Rod Withdrawal Error, or Load Rejection Without Main Turbine Bypass operable) initiated from a reduced power condition.

Insert 17

In addition, the MCPR limits for single-loop operation protect against the effects of the Recirculation Pump Seizure Accident. That is, for operation in single-loop with an operating MCPR limit $\geq$ 1.30, the radiological consequences of a pump seizure accident from single-loop operating conditions are but a small fraction of 10CFR100 guidelines.

Insert 18

One fuel assembly shall contain 78 fuel rods, one inert rod, and 2 water rods. All other fuel assemblies shall contain

Insert 19

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies consisting of two different designs. The "original equipment" design consists of a cruciform array of stainless steel tubes containing 143 inches of boron carbide (B4C) powder surrounded by a stainless steel sheath. The "replacement" control blade design consists of a cruciform array of stainless steel tubes containing 143 inches of boron carbide (B4C) powder near the center of the cruciform, and 143 inch long solid hafnium rods at the edges of the cruciform, all surrounded by a stainless steel sheath.

NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following three questions are addressed for each of the proposed Technical Specification changes:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

3. Does the proposed change involve a significant reduction in a margin of safety?

Specifications 1.0 - Definitions, and 3/4.2.3, Minimum Critical Power Ratio

The changes to these specifications support new MCPR operating limits based on the PP&L reactor analysis methods described in Reload Summary Report Reference 3. The limits calculated for U2C5 will be a function of scram speed. Therefore, the format for Specification 3/4.2.3 has changed significantly and the new definition is required.

1. No. The MCPR operating limits for U2C5 were generated with the PP&L reactor analysis methods described in PL-NF-90-001 (See Reload Summary Report Reference 3). The U2C5 MCPR operating limits are presented as MCPR versus Percent of Rated Core Flow and MCPR versus Percent Core Thermal Power. These limits cover the allowed operating range of power and flow. As specified in PL-NF-90-001, six major events were analyzed. These events can be divided into two categories: core-wide transients and local transients. The core-wide transient events analyzed were:

   1) Generator Load Rejection Without Bypass (GLRWOB),

   2) Feedwater Controller Failure (FWCF),

   3) Recirculation Flow Controller Failure - Increasing Flow (RFCF), and

   4) Loss of Feedwater Heating (LOFWH)

   As discussed in PL-NF-90-001, the other core-wide transients are non-limiting (i.e., would produce lower calculated ΔCPRs than one of the four events analyzed). The local transient events analyzed were:

   1) Rod Withdrawal Error (RWE), and

   2) Fuel Loading Error (FLE).

   The fuel loading error evaluation includes analysis of both rotated and mislocated fuel bundles.

Sufficient analyses were performed to define the MCPR operating limits as a function of core power and core flow. Analyses were also performed to determine MCPR operating limits for three plant equipment availability conditions: 1) Turbine Bypass and EOC-RPT operable, 2) Turbine Bypass inoperable, and 3) EOC-RPT inoperable.

Core-Wide Transients

The PP&L RETRAN model and methods described in PL-NF-89-005 and PL-NF-90-001 (See Reload Summary Report References 2 and 3) were used to analyze the GLRWOB, FWCF, and RFCF events. The ΔCPRs were evaluated using the XN-3 Critical Power Correlation (See Reload Summary Report Reference 26) and the methodology described in PL-NF-90-001 (See Reload Summary Report Reference 3). The GLRWOB and FWCF events were analyzed in two different ways (as described in PL-NF-90-001):

1) Deterministic analyses using the Technical Specification scram speed (minimum allowed);

2) Statistical Combination of Uncertainty (SCU) analyses at an average scram speed of 4.2 feet/second.

Thus, the Technical Specification MCPR operating limits calculated for U2C5 will be a function of scram speed.

The LOFWH event was conservatively analyzed by PP&L using the steady state core physics methods and process described in Reload Summary Report References 1 and 3, and the LOFWH event results were found to be bounded by results of the other three core-wide transients. The minimum MCPR operating limit required for the U2C5 LOFWH event is 1.17.

Results of the GLRWOB, FWCF, and RFCF events are presented in Reload Summary Report Tables 3, 4, and 5, respectively.

Local Transients

The fuel loading error (rotated and mislocated bundle) and the Rod Withdrawal Error (RWE) were analyzed using the methodology described in PL-NF-90-001. The results of these analyses apply to all three plant equipment availability conditions previously described, and the results are independent of scram speed. The RWE analysis supports the use of both the Duralife 160C control blades and a Rod Block Monitor setpoint of 108%. The MCPR operating limits that result from the analyses of these events are presented in Reload Summary Report Table 6. These events are non-limiting for U2C5.

Based on the above, the methodology used to develop the new MCPR operating limits for the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The methodology and results described above can only be evaluated for their effect on the consequences of analyzed events; they cannot create new ones. The consequences of analyzed events were evaluated in 1. above.

3. No. Based on 1. above, the methodology used to generate the MCPR operating limits for U2C5 is both sufficient and conservative. Furthermore, although the methodology (PL-NF-90-001) is still undergoing NRC review, PP&L believes it meets all pertinent regulatory criteria for use in this application. Therefore, its use will not result in a significant decrease in any margin of safety.

<u>Specification 2.1.2 - Thermal Power, High Pressure and High Flow</u>

1. No. The PP&L Statistical Combination of Uncertainties (SCU) methods are described in Reload Summary Report Reference 3. When using the SCU methodology, the transient ΔCPR and traditional MCPR safety limit analyses are combined through a single unified analysis. As a result, the Thermal Power, High Pressure and High Flow safety limit is not represented as a single MCPR value, but rather as a condition such that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. As described in Appendix B of Reload Summary Report Reference 3, this combined analysis and compliance with the resulting safety limit condition are supported by "MCPR Safety Limit type" calculations. The "MCPR Safety Limit type" calculations were performed by ANF using the same methods and assumptions as the traditional MCPR Safety Limit analysis.

   As shown in Reload Summary Report Table 1, a MCPR value of 1.06 in two loop operation assures that less than 0.1% of the fuel rods are expected to experience boiling transition. The methodology and generic uncertainties used in the "MCPR Safety Limit type" calculations are provided in XN-NF-80-19(P)(A), Volume 4 Revision 1 (Reload Summary Report Reference 6). The uncertainties used for the SSES U2C5 "MCPR Safety Limit type" calculations are the same as for U2C4 and are presented in Reload Summary Report Reference 18. The results are presented in Reload Summary Report Table 1.

   During U2C5, as in the previous cycle, the ANF 9x9 fuel will be monitored using the XN-3 critical power correlation. ANF has determined that this correlation provides sufficient conservatism to preclude the need for any penalty due to channel bow during U2C5. Susquehanna SES is a C-lattice plant and uses channels for only one fuel bundle lifetime. The conservatism has been evaluated by ANF to be greater than the maximum expected ΔCPR (0.02) due to channel bow in C-lattice plants using channels for only one fuel bundle lifetime. Therefore, the monitoring of the MCPR limit is conservative with respect to channel bow and addresses the concerns of NRC Bulletin No. 90-02. The details of the evaluation performed by ANF have been reported generically to the NRC (Reload Summary Report Reference 17).

Based on the above, the methodology used to develop the new safety limit condition for the Technical Specification does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The methodology and results described above can only be evaluated for their effect on the consequences of analyzed events; they cannot create new ones. The consequences of analyzed events were evaluated in 1. above.

3. No. Based on 1. above, the methodology used to generate the Thermal Power, High Pressure and High Flow safety limit condition for U2C5 is both sufficient and conservative. Furthermore, although the methodology (PL-NF-90-001) is still undergoing NRC review, PP&L believes it meets all pertinent regulatory criteria for use in this application. Therefore, its use will not result in a significant decrease in any margin of safety.

<u>Specification 3/4.4.1, Recirculation System - Two Loop Operation</u>

The changes to this specification (i.e., Figure 3/4.1.1.1-1) reflect cycle-specific stability analyses.

1. No. COTRAN core stability calculations were performed for Unit 2 Cycle 5 to determine the decay ratios at predetermined power/flow conditions. The resulting decay ratios were used to define operating regions which comply with the interim requirements of NRC Bulletin No. 88-07, Supplement 1 "Power Oscillations in Boiling Water Reactors". As in the previous cycle, Regions B and C of the NRC Bulletin have been combined into a single region (i.e., Region II), and Region A of the NRC Bulletin corresponds to Region I.

   Region I has been defined such that the decay ratio for all allowable power/flow conditions outside of the region is less than 0.90. To mitigate or prevent the consequences of instability, entry into this region requires a manual reactor scram. Region I for Unit 2 Cycle 5 is slightly different than Region I for the previous cycle.

   Region II has been defined such that the decay ratio for all allowable power/flow conditions outside of the region (excluding Region I) is less than 0.75. For Unit 2 Cycle 5, Region II must be immediately exited if it is inadvertently entered. Similar to Region I, Region II is slightly different than in the previous cycle.

   In addition to the region definitions, PP&L has performed stability tests in SSES Unit 2 during initial startup of Cycles 2, 3 and 4 to demonstrate stable reactor operation with ANF 9x9 fuel. The test results for U2C2 (See Reload Summary Report Reference 20) show very low decay ratios with a core containing 324 ANF 9x9 fuel assemblies.

Figure 3/4.1.1.1-1 is also referenced by Specification 3/4.4.1.1.2, which governs Single Loop Operation (SLO). The evaluation above applies under SLO conditions as well.

Based on the above, operation within the limits specified by the proposed changes will ensure that the probability and consequences of unstable operation will not significantly increase.

2. No. The methodology described above can only be evaluated for its effect on the consequences of unstable operation; it cannot create new events. The consequences were evaluated in 1. above.

3. No. PP&L believes that the use of Technical Specifications that comply with NRC Bulletin 88-07, Supplement 1, and the tests and analyses described above, will provide assurance that SSES Unit 2 Cycle 5 will comply with General Design Criteria 12, Suppression of Reactor Power Oscillations. This approach is consistent with the SSES Unit 2 Cycle 4 method for addressing core stability (See Reload Summary Report References 4 and 5).

Specification 3/4.4.1, Recirculation System - Single Loop Operation

The changes to this specification are either evaluated above or are editorial in nature. The reference to Specification 2.1.2 is deleted because the new limit (see Evaluation of Specification 2.1.2 above) will not change for Single Loop Operation. The additional figures referenced from Specification 3.2.3 are the result of the MCPR operating limit analyses evaluated above.

The other two changes to Surveillance Requirements 4.4.1.1.2.6, correct inadvertent typographical errors that occurred during the issuance of Amendment 60 to the Unit 2 Technical Specifications.

1. No. The changes are either evaluated elsewhere in this No Significant Hazards Considerations evaluation, or are entirely editorial in nature.

2. No. See 1. above.

3. No. See 1. above.

Specification 5.3.1 - Fuel Assemblies

This section has been changed to describe the actual core configuration for U2C5, which includes one inert (i.e., solid zircaloy-2) rod.

1. No. The inert rod was used to repair a fuel assembly that failed during U2C2. This repaired assembly was analyzed and found to be acceptable in support of U2C4 operation, which was approved by the NRC (See Reload Summary Report Reference 5). Based on the above, use of the repaired assembly does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. See I above.

3. No. See I above.

Specification 5.3.2 - Control Rod Assemblies

The changes to this specification are provided in order to recognize the replacement control blade design being utilized in U2C5.

1. No. The main differences between the replacement Duralife 160C control blades and the original equipment control blades are:

   a) the Duralife 160C control blades utilize three solid hafnium rods at each edge of the cruciform to replace the three B_4C rods that are most susceptible to cracking and to increase control blade life;

   b) the Duralife 160C control blades utilize improved B_4C tube material (i.e. high purity stainless steel vs. commercial purity stainless steel) to eliminate cracking in the remaining B_4C rods during the lifetime of the control blade;

   c) the Duralife 160C control blades utilize GE's crevice-free structure design, which includes additional B_4C tubes in place of the stiffeners, an increased sheath thickness, a full length weld to attach the handle and velocity limiter, and additional coolant holes at the top and bottom of the sheath;

   d) the Duralife 160C control blades utilize low cobalt-bearing pin and roller materials in place of stellite which was previously utilized;

   e) the Duralife 160C control blade handles are longer by approximately 3.1 inches in order to facilitate fuel moves within the reactor vessel during refueling outages at Susquehanna SES; and

   f) the Duralife 160C control blades are approximately 16 pounds heavier as a result of the design changes described above.

   The Duralife 160C control blade has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and all other lateral and vertical loads expected for each condition. The Duralife 160C control blade stresses, strains, and cumulative fatigue have been evaluated and result in an acceptable margin to safety. The control blade insertion capability has been evaluated and found to be acceptable during all modes of plant operation within the

limits of plant analyses. The Duralife 160C control blade coupling mechanism is equivalent to the original equipment coupling mechanism, and is therefore fully compatible with the existing control rod drives in the plant. In addition, the materials used in the Duralife 160C are compatible with the reactor environment. The impact of the increased weight of the control blades on the seismic and hydrodynamic load evaluation of the reactor vessel and internals has been evaluated and found to be negligible.

With the exception of the crevice-free structure and the extended handle, the Duralife 160C blades are equivalent to the NRC approved Hybrid I Control Blade Assembly (See Reload Summary Report Reference 9). The mechanical aspects of the crevice-free structure were approved by the NRC for all control blade designs in Reload Summary Report Reference 10. A neutronics evaluation of the crevice-free structure for the Duralife 160C design was performed by GE using the same methodology as was used for the Hybrid I control blades in Reload Summary Report Reference 9. These calculations were performed for the original equipment control blades and the Duralife 160C control blades described above assuming an infinite array of ANF 9x9 fuel. The Duralife 160C control blade has a slightly higher worth than the original equipment design, but the increase in worth is within the criterion for nuclear interchangeability. The increase in blade worth has been taken into account in the appropriate U2C5 analyses. However, as stated in Reload Summary Report Reference 9, the current practice in the lattice physics methods is to model the original equipment all B_4C control blade as non-depleted. The effects of control blade depletion on core neutronics during a cycle are small and are inherently taken into account by the generation of a target k-effective for each cycle. As discussed above, the neutronics calculations of the crevice-free structure, show that the non-depleted Duralife 160C control blade has direct nuclear interchangeability with the non-depleted original equipment all B_4C design. The Duralife 160C also has the same end-of-life reactivity worth reduction limit as the all B_4C design. Therefore, the Duralife 160C can be used without changing the current lattice physics model as previously approved for the Hybrid I control blades (Reload Summary Report Reference 9).

The extended handle and the crevice-free structure features of the Duralife 160C control blades result in a one pound increase in the control blade weight over that of the Hybrid I blades, and a sixteen pound increase over the Susquehanna SES original equipment control blades. In Reload Summary Report Reference 9, the NRC approved the Hybrid I control blade which weighs less (by more than one pound) than the D lattice control blade. The basis of the Control Rod Drop Accident analysis continues to be conservative with respect to control rod drop speed since the Duralife 160C control blade weighs less than the D lattice control blades, and the heavier D lattice control blade speed is used in

the analysis. In addition, GE performed scram time analyses and determined that the Duralife 160C control blade scram times are not significantly different than the original equipment control blade scram times. The current Susquehanna SES measured scram times also have considerable margin to the Technical Specification limits. Since the increase in weight of the Duralife 160C control blades does not significantly increase the measured scram speeds and the safety analyses which involve reactor scrams utilize either the Technical Specification limit scram times or a range of scram times up to and including the Technical Specification scram times, the operating limits are applicable to U2C5 with Duralife 160C control blades.

Since the Duralife 160C control blades contain solid hafnium rods in locations where the B_4C tubes have failed, and the remaining B_4C rods are manufactured with an improved tubing material (high purity stainless steel vs commercial purity stainless steel), boron loss due to cracking is not expected. Therefore, the requirements of IE Bulletin 79-26, Revision 1 do not apply to the Duralife 160C control blades. However, PP&L plans to continue tracking the depletion of each control blade and discharge any control blade prior to a ten percent loss in reactivity worth.

Based on the discussion above, the new control blades proposed to be utilized in U2C5 do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The replacement blades can only be evaluated for their effectiveness as part of the overall reactivity control system, which is evaluated in terms of analytical consequences in 1. above. Since they do not cause any significant change in system operation or function, no new events are created.

3. No. The analyses described in 1. above indicate that the replacement blades meet all pertinent regulatory criteria for use in this application, and are expected to eliminate the boron loss concerns expressed in IE Bulletin 79-26, Revision 1. Therefore, the proposed change does not result in a significant decrease in any margin of safety.