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 million lbm/hr.

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 million lbm/hr., 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*~~ 1.09* with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION:

With MCPR less than ~~1.06*~~ 1.09* and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10 million lbm/hr., 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

SPC expected

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 stepback approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specifications 2.1.2 for ~~SNP~~ 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 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~~). 2

ANF-524(P)(A)

2.1.1 THERMAL POWER, Low Pressure or Low Flow

585 ANFB 0.1

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~~ x 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 ~~SNP 9x9~~ fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For the ~~SNP 9x9~~ design, the coolant minimum flow and maximum flow area is such that the mass flux is always greater than 0.25 x 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 x 10^6 lbs/hr-ft^2 is 3.35 Mwt or greater. At

SPC 9x9-2

ANF-[illegible]24(P)(A), Revision 2

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).

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 statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the ~~XN-3~~ critical power correlation. ~~XN-NF-524 (A) Revision 1~~ -A and PL-NF-90-001 describe the methodologies used in determining the Safety Limit MCPR.

The ANFB ~~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 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~~ ANFB 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. 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.

~~SNP~~ SPC fuel is monitored using the ANFB ~~XN-3~~ critical power correlation. ~~SNP 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 SNP 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. Since Susquehanna SES is a C-lattice plant and uses channels for only one fuel bundle lifetime, monitoring of the MCPR limit with the XN-3 critical power correlation is conservative with respect to channel bow and~~ INSERT 1 addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

INSERT 1

The effects of channel bow on MCPR are explicitly included in the calculation of the ANFB MCPR Safety Limit. Explicit treatment of channel bow in the ANFB MCPR Safety Limit ...

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~~ 1.10.
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%

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

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

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the applicable Single Loop Operation MCPR limit as specified in the CORE OPERATING LIMITS REPORT.
5. Specification 3.2.4: The LINEAR HEAT GENERATION RATE (LHGR) shall be less than or equal to the applicable Single Loop Operation LHGR limit as specified in the CORE OPERATING LIMITS REPORT.
6. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ 0.63W + 35%	≤ 0.63W + 37%
	Trip Setpoint	Allowable Value
b. APRM - Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

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

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.

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). 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 ≥ 1.30,~~ the radiological consequences of a pump seizure accident from single-loop operating conditions are ~~but~~ a small fraction of 10 CFR 100 guidelines.

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 on 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.

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.

CORE OPERATING LIMITS REPORT (Continued)

9. XN-NF-84-97, Revision 0, "LOCA-Seismic Structural Response of an ENC 9x9 Jet Pump Fuel Assembly," Exxon Nuclear Company, Inc., December 1984.

10. PLA-2728, "Response to NRC Question : Seismic/LOCA Analysis of U2C2 Reload," Letter from H.W. Keiser (PP&L) to E. Adensam (NRC), September 25, 1986.

11. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 Fuel," May 1988.

INSERT 2

12. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements ~~1 and 2~~ 1, 2, and 3, "Exxon Nuclear Methodology for Boiling Water Reactors : Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc., March 1983.

~~13. XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc., November 1983.~~

~~14. XN-NF-512-P-A, Revision 1 and Supplement 1, Revision 1, "XN-3 Critical Power Correlation," October, 1982.~~

15. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

16. NE-092-001A, Revision 1, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992.

17. NRC SER on PP&L Power Uprate LTR (November 30, 1993).

18. PL-NF-90-001, Supplement 1-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Loss of Feedwater Heating Changes and Use of RETRAN MOD 5.1," September 1994.

19. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES", January, 1995.

6.9.3.3 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.

INSERT 3

INSERT 2

13. ANF-524(P)(A), Revision 2 and Supplement 1, Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors", November 1990.

14. ANF-1125(P)(A) and ANF-1125 (P)(A), Supplement 1, "ANFB Critical Power Correlation", April 1990.

INSERT 3

20. CENPD-300-P, "Reference Safety Report for Boiling Water Reactor Reload Fuel", ABB Combustion Engineering Nuclear Operations, November 1994.

21. PL-NF-90-001, Supplement 2, "Application of Reactor Analysis Methods for BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation".