## 2.0 SAFETY LIMITS and LIMITING SAFETY SYSTEM SETTINGS

1.07 UP TO 4500 MWD/MTU CYCLE EXPOSURE AND 1.11 FOR CYCLE EXPOSURE GREATER THAN 4500 MWD/MTU TO EOC

BASES

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 ~~1.06~~ for two recirculation loop operation and ~~1.07~~ for single recirculation loop operation for all nuclear fuel in WNP-2. MCPR greater than ~~1.06~~ for two recirculation loop operation and ~~1.07~~ for single recirculation loop operation 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 integrity Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. ANF 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 percent of the fuel rods in the core do not experience (P) transition boiling (Reference: XN-NF-524(A), Rev. 2 ~~1~~; ABB Atom Report UK90-126; GE11 Lead Fuel Assembly Report for Washington Public Power Supply System Nuclear Project No. 2, Reload 5, Cycle 6). The latter two references support application of the above established safety limit to GE11 and SVEA-96 LFA fuel in WNP-2.

## 2.1 SAFETY LIMITS

1.08 UP TO 4500 MWD/MTU CYCLE EXPOSURE AND 1.12 FOR CYCLE EXPOSURE GREATER THAN 4500 MWD/MTU TO EOC

### 2.1.1. THERMAL POWER, Low Pressure or Low Flow

0.10 ANFB ANF B

For certain conditions of pressure and flow, the ~~XN-3~~ correlation is not valid for all critical power calculations. The ~~XN-3~~ correlation is not valid for bundle mass velocities less than ~~.25~~ x $10^6$ lbs/hr-ft$^2$ or pressures less than ~~585 psig~~ 590 psia. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x $10^3$ lbs/h (approximately a mass velocity of ~~.25~~ x $10^6$ lbs/hr-ft$^2$), bundle pressure drop is nearly independent of bundle power

0.25

THERMAL POWER, Low Pressure or Low Flow (Continued)

and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x $10^3$ lbs/h. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below ~~585 psig~~ 785 is conservative.

## 2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors[(a)] which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the ANF nuclear critical heat fluxenthalpy ANFB ~~XN-3~~ correlation. The ~~XN-3~~ ANFB correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1.

The bases for the reactor system and fuel uncertainties ~~in the core parameters~~ are given in ANF XN-NF-524(A)(P), Rev. ~~1~~ 2[(a)] ~~. and the basis for the uncertainty in the XN-3 correlation is given in XN-NF-512(A), Rev. 1[(b)]~~ ~~ANF B~~ ~~ANF-1125(P)(A) and SUPPLEMENTS 1 AND 2~~. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

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a. ~~Exxon~~ Advanced Nuclear FUELS Critical Power Methodology for Boiling Water Reactors, ANF XN-NF-524(A)(P), Rev. ~~1~~ 2

~~b. Exxon Nuclear Company XN-3 Critical Power Correlation, XN-NF-512(A), Rev. 1.~~

~~ANFB CRITICAL POWER CORRELATION, ANF-1125(P)(A) and SUPPLEMENTS~~ 1 and 2.

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BASES TABLE B2.1.2-1

UNCERTAINTIES CONSIDERED IN

THE MCPR SAFETY LIMIT

| Parameter | STANDARD DEVIATION* |
|---|---|
| Feedwater Flow Rate | .0176 |
| Feedwater Temperature | .0076 |
| Core Pressure | .0050 |
| Total Core Flow Rate | .0250 |
| ~~Core Inlet Enthalpy~~ | ~~.0024~~ |
| ANFB ~~XN-3~~ Critical Power Correlation | .0310 ~~.0411~~ |
| Assembly Flow Rate | .0280 |
| Power Distribution: | |
| Radial ~~Peaking Factor~~ ASSEMBLY POWER | ~~.0528~~ ~~.0307~~ .0409 |
| Local ~~Peaking Factor~~ POWER** | ~~.0246~~ .0229 |
| ANFB CORRELATION Additive CONSTANTS | .0200 |

delete

* Fraction of Nominal Value.

** RELATIVE LOCAL ROD POWER