ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

MODIFIED SAFETY LIMIT

THERMAL POWER, HIGH PRESSURE AND HIGH FLOW

SUMMARY JUSTIFICATION FOR

TECHNICAL SPECIFICATION CHANGES

TECHNICAL SPECIFICATION NO.	PAGE NO.	JUSTIFICATION
2.1.2	2-1	New safety limit values to reflect cycle specific safety analysis, employing new methodology.
B2.0	B2-1	Editorial change to bases to reflect change to safety limit 2.1.2 discussed above.

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.

1.07 up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than ~~1.06~~ with two recirculation loop operation and shall not be less than ~~1.07~~ with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

1.08 up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION: 1.07 up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC

With MCPR less than ~~1.06~~ with two recirculation loop operation or less than ~~1.07~~ with single recirculation loop operation 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.

1.08 up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC

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.

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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. 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 transition boiling (Reference: XN-NF-524(A), Rev. 2X; 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. (P)

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 psig. 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 psig 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~~ correlation is valid over the range of conditions used in the tests of the ANF B 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 uncertainties in the core parameters are given in XN-NF-524(A), Rev. ~~1~~ 2[(a)] and the basis for the uncertainty in the ~~XN-3~~ ANF B correlation is given in ~~XN-NF-512(A), Rev. 1~~ ANF-1125(P)(A) and SUPPLEMENTS 1 and 2.[(b)] 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, XN-NF-524(A), 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 ASSEMBLY POWER ~~Peaking Factor~~	~~.0528~~ .0301
Local ~~Peaking Factor~~ POWER**	~~.0246~~ .0229
ANFB CORRELATION Additive CONSTANTS	.02

* Fraction of Nominal Value.

** RELATIVE LOCAL ROD POWER

APPENDIX A
MCPR SAFETY LIMIT

A.1 INTRODUCTION

Bundle power limits in a boiling water reactor (BWR) are determined through evaluation of critical heat flux phenomena. The basic criterion used in establishing critical power ratio (CPR) limits is that at least 99.9% of the fuel rods in the core will be expected to avoid boiling transition (critical heat flux) during normal operation and anticipated operational occurrences. Operating margins are defined by establishing a minimum margin to the onset of boiling transition condition for steady state operation and calculating a transient effects allowance, thereby assuring that the steady state limit is protected during anticipated off-normal conditions. This appendix addresses the calculation of the minimum margin to the steady state boiling transition condition, which is implemented as the MCPR safety limit in the plant technical specifications. The transient effects allowance, or the limiting transient change in CPR (i.e., ΔCPR), is treated in the body of this report.

The MCPR safety limit is established through statistical consideration of measurement and calculational uncertainties associated with the thermal hydraulic state of the reactor using design basis radial, axial, and local power distributions and considering fuel assembly channel bow. Reference 5 describes ANF MCPR safety limit methodology and the incorporation of channel bow effects. Some of the calculational uncertainties, including those introduced by the critical power correlation, power peaking, and core coolant distribution, are fuel related. When ANF fuel is introduced into a core where it will reside with another supplier's fuel types, the appropriate value of the MCPR safety limit is calculated based on fuel-dependent parameters associated with the mixed core. Similarly, when an ANF-fabricated reload batch is used to replace a group of dissimilar fuel assemblies, the core average fuel dependent parameters change because of the difference in the relative number of each type of bundle in the core, and the MCPR safety limit is again reevaluated.

The design basis power distribution is made up of components corresponding to representative radial, axial, and local peaking factors. Where such data are appropriately available from the previous cycle, these factors are determined through examination of operating data for the previous cycle and predictions of operating conditions during the cycle being evaluated for the MCPR safety limit. Available operating data for WNP-2 and the predicted operating conditions for Cycle 7 were evaluated to identify the design basis power distributions for use in the Cycle 7 MCPR safety limit analysis. A high neutron flux trip of 120% was used in the safety limit analysis.

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A.2 ASSUMPTIONS

A.2.1 Design Basis Power Distribution

The local and radial power distributions which were determined to be conservative for use in the safety limit analysis are shown in Figures A.1 through A.4.

A.2.2 Hydraulic Demand Curve

Hydraulic demand curves based on calculations with XCOBRA were used in the safety limit analysis. The XCOBRA calculation is described in ANF topical reports XN-NF-79-59(A), "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," and ANF-1125(P)(A), "ANFB critical Power Correlation."

A.2.3 System Uncertainties

System measurement uncertainties are not fuel dependent. The values reported by the NSSS supplier for these parameters remain valid for the insertion of ANF fuel. The values used in the safety limit analysis are tabulated in the topical report XN-NF-524(P)(A), Revision 2 and Supplements, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors."

A.2.4 Fuel Related Uncertainties

Fuel related uncertainties include power measurement uncertainty and core flow distribution uncertainty. The values used in the safety limit analysis are also tabulated in the topical report XN-NF-524(P)(A), Revision 2 and Supplements, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors."

A.3 SAFETY LIMIT CALCULATION

A statistical analysis for the number of fuel rods in boiling transition was performed using the methodology described in ANF topical report XN-NF-524(P)(A), Revision 2 and Supplements, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors." With 500 Monte Carlo trials it was determined that for a minimum CPR value of 1.07 at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition with a confidence level of 95% for the design basis power distributions from BOC to a cycle average burnup of 4500 MWd/MTU. Similarly for the design basis power distributions from 4500 MWd/MTU to EOC a minimum CPR value of 1.11 is required to provide the same protection.

The effects of channel bow are included in the 1.07 and 1.11 safety limit MCPR values. Without channel bow, the SLMCPRs would have been reduced by about 0.03. The Supply System has reused some initial core channels on ANF 8x8 fuel assemblies in the WNP-2 Cycle 7 core. A maximum extent of channel bow including the effects of reused channels adjacent to assemblies with exposed channels was input for the 8x8 fuel. The 9x9-9X fuel input was based on a maximum extent of channel bow assuming a new channel on the 9x9-9X fuel adjacent to assemblies with exposed channels. The core results including input for both fuel types show that the ANF 8x8 assemblies having reused channels in the WNP-2 Cycle 7 core have sufficient MCPR margin that they do not contribute significantly to the number of rods in boiling transition even with the effects of increased channel bow due to reused channels. The maximum end of cycle exposure of the reused channels corresponds to a bundle average burnup of 48,000 MWd/MTU which is within the ANF channel bow data base to approximately 50,000 MWd/MTU.

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1.026	1.063	1.062	1.023	1.125	1.012	1.036	1.020	0.966
1.063	0.959	1.043	1.021	0.870	1.009	1.016	0.919	0.996
1.062	1.043	1.021	1.080	1.109	1.067	0.995	0.996	0.991
1.023	1.021	1.080	0.000	0.000	0.000	1.052	0.976	0.954
1.125	0.870	1.109	0.000	0.000	0.000	1.081	0.834	1.048
1.012	1.009	1.067	0.000	0.000	0.000	1.042	0.967	0.944
1.036	1.016	0.995	1.052	1.081	1.042	0.975	0.875	0.968
1.020	0.919	0.996	0.976	0.834	0.967	0.875	0.884	0.955
0.966	0.996	0.991	0.954	1.048	0.944	0.968	0.955	0.907

FIGURE A.1 WNP-2 CYCLE 7 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF 9X9-9X FUEL WITH CHANNEL BOW)

1.047	1.054	1.096	1.081	1.068	1.085	1.100	1.075
1.054	1.045	0.940	1.060	1.042	0.934	0.950	1.060
1.096	0.940	1.037	1.009	0.977	0.979	0.906	1.016
1.081	1.060	1.009	0.000	0.883	0.953	0.987	0.978
1.068	1.042	0.977	0.883	0.000	0.963	0.982	0.971
1.085	0.934	0.979	0.953	0.963	0.967	0.855	0.991
1.100	0.950	0.906	0.987	0.982	0.855	0.906	1.007
1.075	1.060	1.016	0.978	0.971	0.991	1.007	0.993

FIGURE A.2 WNP-2 CYCLE 7 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF 8X8 FUEL WITH REUSED CHANNEL BOW)

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1.047	1.054	1.096	1.081	1.068	1.085	1.100	1.075
1.054	1.045	0.940	1.060	1.042	0.934	0.950	1.060
1.096	0.940	1.037	1.009	0.977	0.979	0.906	1.016
1.081	1.060	1.009	0.000	0.883	0.953	0.987	0.978
1.068	1.042	0.977	0.883	0.000	0.963	0.982	0.971
1.085	0.934	0.979	0.953	0.963	0.967	0.855	0.991
1.100	0.950	0.906	0.987	0.982	0.855	0.906	1.007
1.075	1.060	1.016	0.978	0.971	0.991	1.007	0.993

FIGURE A.2 WNP-2 CYCLE 7 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF 8X8 FUEL WITH REUSED CHANNEL BOW)

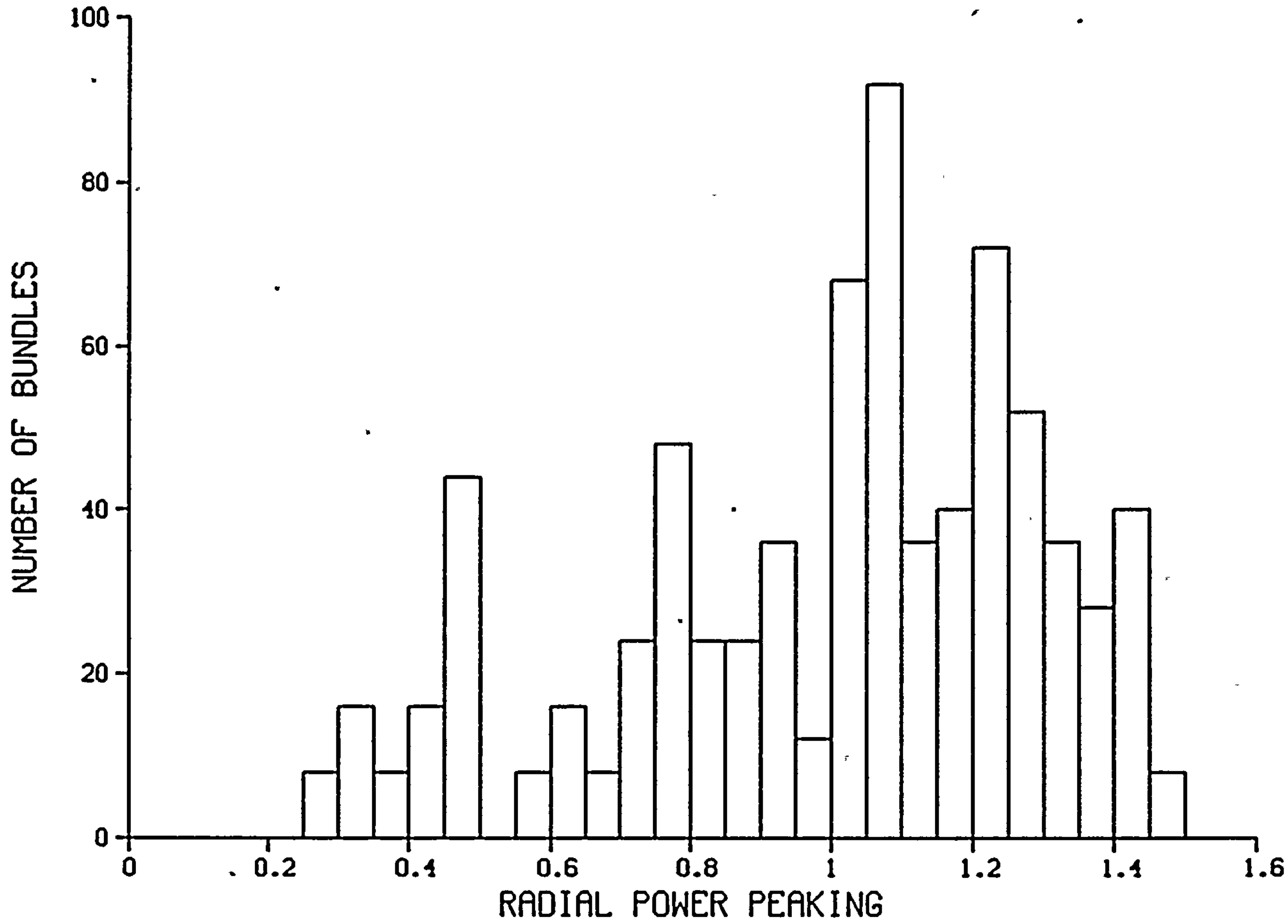

FIGURE A.3 RADIAL POWER HISTOGRAM FOR FULL CORE SAFETY LIMIT MODEL
BOL TO 4500 MWd/MTU CYCLE AVERAGE BURNUP

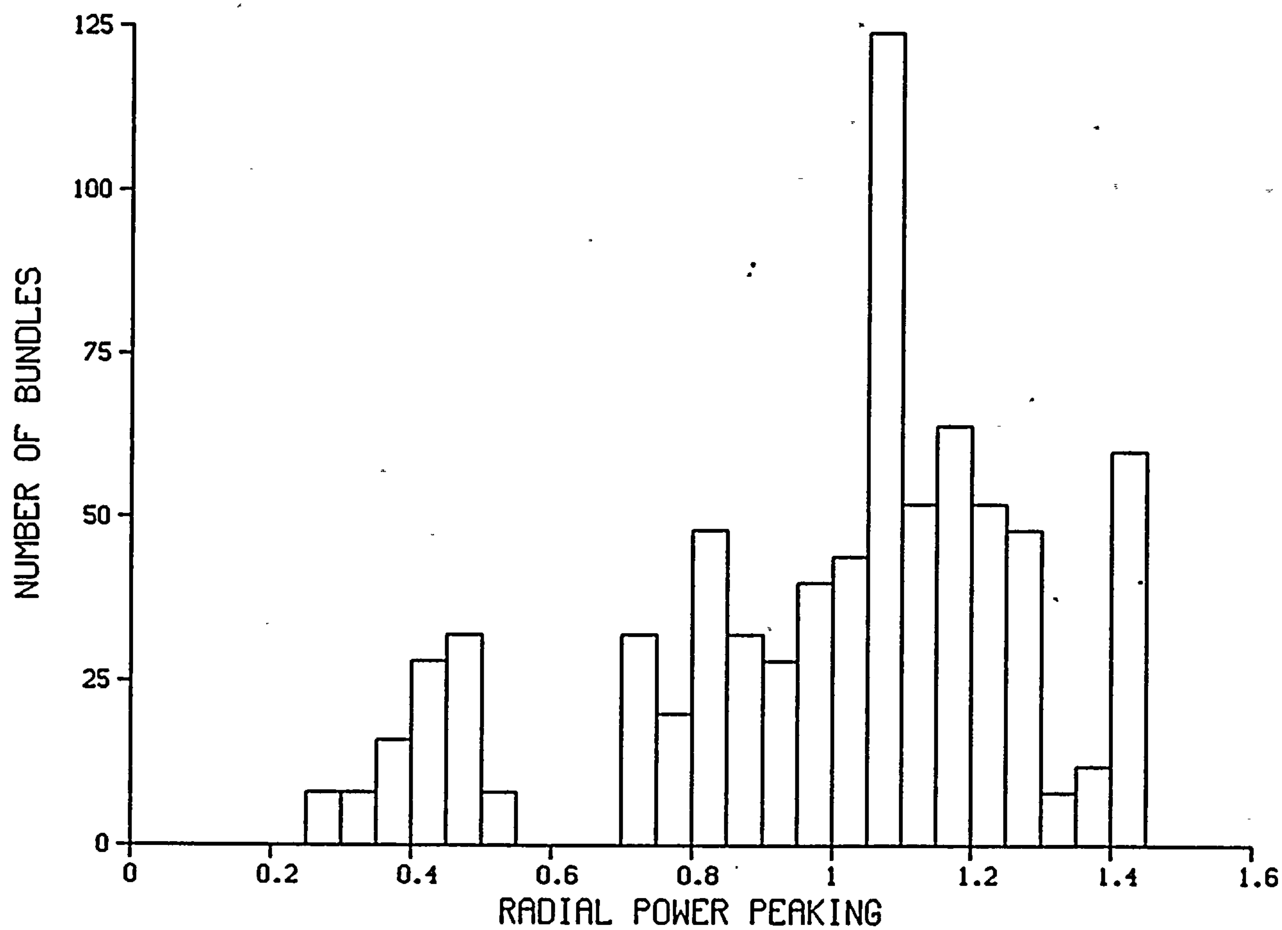

FIGURE A.4 RADIAL POWER HISTOGRAM FOR FULL CORE SAFETY LIMIT MODEL
4500 MWd/MTU CYCLE AVERAGE BURNUP TO EOC

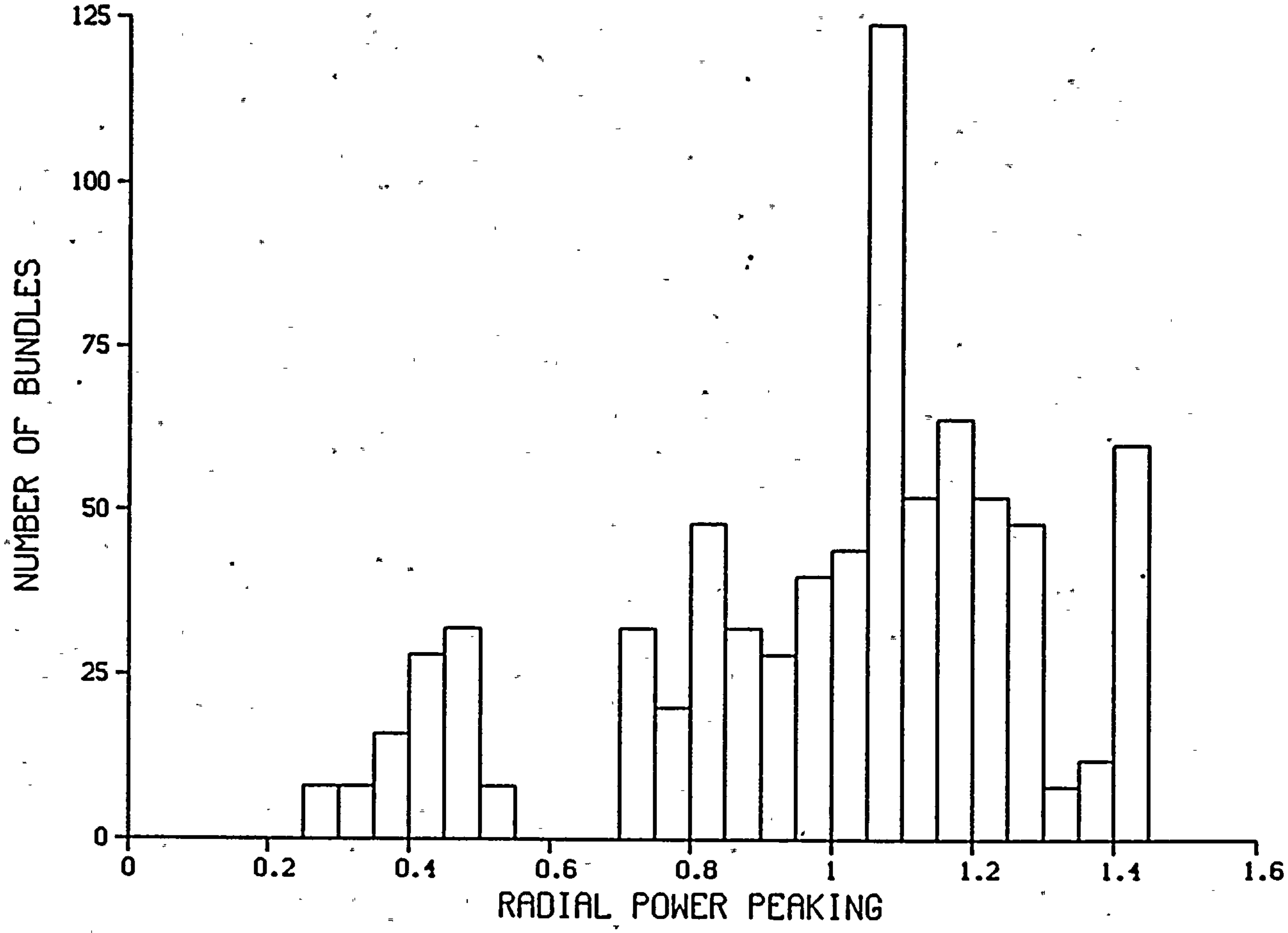

FIGURE A.4 RADIAL POWER HISTOGRAM FOR FULL CORE SAFETY LIMIT MODEL
4500 MWd/MTU CYCLE AVERAGE BURNUP TO EOC