ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES

SUMMARY JUSTIFICATION FOR

TECHNICAL SPECIFICATION CHANGES

TECHNICAL SPECIFICATION NO.	PAGE NO.	JUSTIFICATION
Index	xx(a)	Previous figure change.
2.1.2	2-1	New safety limit values to reflect cycle specific safety analysis.
B2.0	B2-1	Change to bases to reflect change to safety limit 2.1.2 discussed above.
6.9.3.2	6-21	Changes to reference document titles and report status.

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LIST OF FIGURES

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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 1.07 ~~up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC~~ with two recirculation loop operation and shall not be less than 1.08 ~~up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC~~ 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.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 ~~up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC~~ with two recirculation loop operation or less than 1.08 ~~up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC~~ 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.

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.

2.0 SAFETY LIMITS and LIMITING SAFETY SYSTEM SETTINGS

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 1.07 ~~up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC~~ for two recirculation loop operation and 1.08 ~~up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC~~ for single recirculation loop operation for all nuclear fuel in WNP-2. MCPR greater than 1.07 ~~up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC~~ for two recirculation loop operation and 1.08 ~~up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC~~ 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: ANF-524(P)(A), Rev. 2; 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

2.1.1. THERMAL POWER, Low Pressure or Low Flow

For certain conditions of pressure and flow, the ANFB correlation is not valid for all critical power calculations. The ANFB correlation is not valid for bundle mass velocities less than 0.10×10^6 lbs/hr-ft^2 or pressures less than 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

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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
Assembly Flow Rate	.0280
Power Distribution:	
Radial Assembly Power	.0409
Local Power**	.0229
ANFB Correlation Additive Constants	~~.0200~~
8X8 FUEL	.0200
9X9-2 FUEL	.0200
9X9-9X FUEL	.0080

*Fraction of Nominal Value.
**Relative Local Rod Power.

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CORE OPERATING LIMITS REPORT (Continued)

6.9.3.2 The analytical methods used to determine the core operating limits shall be those topical reports and those revisions and/or supplements of the topical report previously reviewed and approved by the NRC, which describe the methodology applicable to the current cycle. For WNP-2 the topical reports are:

1. ANF-1125(P)(A), and Supplements 1 and 2, "ANFB Critical Power Correlation," April 1990

2. Letter, R. C. Jones (NRC) to R. A. Copeland (ANF), "NRC Approval of ANFB Additive Constants for ANF 9x9-9X BWR Fuel," dated November 14, 1990

3. ~~XN~~ ANF-NF-524(P)(A), Revision 2 and Supplements 1 and 2, "~~Exxon Nuclear~~ ADVANCED NUCLEAR FUELS CORPORATION Critical Power Methodology for Boiling Water Reactors," November 1990

4. ANF-913(P)(A), Volume 1, Revision 1 and Volume 1, Supplements 2, 3 and 4, "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis," August 1990

5. ANF-CC-33(P)(A), Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option," January 1991

6. XN-NF-80-19(P)(A), Volume 1, Supplements 3 and 4, "~~Exxon Nuclear~~ ADVANCED NUCLER FUELS Methodology for Boiling Water Reactors," November 1990

7. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology ~~for~~ Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," June 1986

8. XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," January 1987

9. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump ~~Boiling Water Reactor~~ BWR Reload Fuel," September 1986

10. ANF-89-014(P)(A), REVISION 1 and SUPPLEMENTS 1 AND 2, "ADVANCED NUCLEAR FUELS CORPORATION ~~"~~Generic Mechanical Design for ~~ANF~~ ADVANCED NUCLEAR FUELS 9x9-IX and 9x9-9X BWR Reload Fuel," ~~May 1989~~ OCTOBER 1991.

~~11. ANF-89-014(P), Supplement 1, "Generic Mechanical Design of ANF 9x9-IX and 9x9-9X BWR Reload Fuel," June 1990~~

~~12. Letter, A.C. Thadani (NRC) to R. Copeland (ANF) "Acceptance for Referencing of Topical Report ANF-89-014(P), Rev. 0, Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel," April 22, 1991~~

11. ~~13.~~ XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology," November 1983

12. ~~14.~~ NEDE-24011-P-A-6, "General Electric Standard Application for Reactor Fuel," April 1983

Attachment 2

MCPR SAFETY LIMIT
APPENDIX A

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 letter report 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), will be addressed in the WNP-2 Cycle 8 Plant Transient Analysis 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 1 describes SNP 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 SNP 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 SNP-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. Available operating data for Cycle 7 was used to develop the predicted operating conditions for Cycle 8. The predicted operating

conditions for Cycle 8 were evaluated to identify the design basis power distributions for use in the Cycle 8 MCPR safety limit analysis. A high neutron flux trip of 126.2% was used in the safety limit analysis.

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.3. The axial power distribution used in the safety limit analysis is not shown because it does not have a significant effect on the results of the analysis.

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 Reference 2.

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 SNP fuel. The values used in the safety limit analysis are tabulated in Table 5.1 of Reference 1.

A.2.4 Fuel Related Uncertainties

Fuel related uncertainties include power measurement uncertainty and core flow distribution uncertainty. Fuel related uncertainties are also tabulated in Table 5.1 of Reference 1. The radial bundle power uncertainty shown in Table 5.1 of Reference 1 was not used in the safety analyses. Instead, as required by the USNRC SER for Reference 3, a more conservative radial power uncertainty based on a TIP asymmetry uncertainty of 6.0 percent was used in the safety limit analysis. In addition to the uncertainty for local power shown in Table 5.1 of Reference 1, the uncertainty in pin power due to channel bow which is described in Supplement 1 of Reference 1 was also used in the safety limit analyses.

A.2.5 Critical Power Correlation Uncertainties

The uncertainty in the ANFB critical power correlation is accounted for as an uncertainty in the local peaking additive constant. The ANFB correlation additive constant for several SNP fuel designs are shown in Table 6.2 of Reference 4 Supplement 1. The ANFB correlation additive constant uncertainty shown in Table 5.1 of Reference 1 corresponds to

SNP 8x8 and 9x9-2 fuel and was used in the safety limit analyses because it is conservatively applicable to the SNP 9x9-9X reload fuel.

A.3 SAFETY LIMIT CALCULATION

A statistical analysis for the number of fuel rods in boiling transition was performed using the methodology described in Reference 1. With 250 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 WNP-2 Cycle 8 design basis power distributions.

The effects of channel bow are included in the 1.07 safety limit MCPR value. Local power distributions with and without channel bow were calculated for the SNP 9x9-9X reload fuel. The calculations with channel bow conservatively assume the assembly is surrounded by highly exposed assemblies. The assembly power of SNP 8x8 fuel is low enough that these assemblies did not contribute to the number of rods in boiling transition.

A.4 REFERENCES

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

2. "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," XN-NF-79-59(P)(A), Exxon Nuclear Company, Richland WA, November 1983.

3. "Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," XN-NF-80-19(P)(A) Volume 1 Supplement 3, Advanced Nuclear Fuels Corporation, Richland, WA, November 1990.

4. "ANFB Critical Power Correlation," ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, Richland, WA, April 1990.

1.005	1.044	1.045	1.011	1.113	1.003	1.027	1.015	0.963
1.044	0.955	1.031	1.013	0.876	1.005	1.013	0.927	0.996
1.045	1.031	1.012	1.072	1.104	1.064	0.994	0.998	0.995
1.011	1.013	1.072	0.000	0.000	0.000	1.053	0.981	0.962
1.113	0.876	1.104	0.000	0.000	0.000	1.085	0.850	1.058
1.003	1.005	1.064	0.000	0.000	0.000	1.047	0.976	0.955
1.027	1.013	0.994	1.053	1.085	1.047	0.982	0.885	0.981
1.015	0.927	0.998	0.981	0.850	0.976	0.885	0.904	0.970
0.963	0.996	0.995	0.962	1.058	0.955	0.981	0.970	0.923

FIGURE A.1 WNP-2 CYCLE 8 SAFETY LIMIT LOCAL PEAKING FACTORS
(SNP 9X9-9X FUEL LOADED IN CYCLE 8 WITH CHANNEL BOW)

1.007	1.036	1.042	1.017	1.105	1.008	1.024	1.006	0.965
1.036	0.966	1.029	1.015	0.907	1.007	1.010	0.937	0.988
1.042	1.029	1.017	1.066	1.092	1.057	0.999	0.996	0.991
1.017	1.015	1.066	0.000	0.000	0.000	1.046	0.983	0.966
1.105	0.907	1.092	0.000	0.000	0.000	1.072	0.878	1.050
1.008	1.007	1.057	0.000	0.000	0.000	1.039	0.976	0.959
1.024	1.010	0.999	1.046	1.072	1.039	0.983	0.893	0.975
1.006	0.937	0.996	0.983	0.878	0.976	0.893	0.910	0.960
0.965	0.988	0.991	0.966	1.050	0.959	0.975	0.960	0.924

FIGURE A.2 WNP-2 CYCLE 8 SAFETY LIMIT LOCAL PEAKING FACTORS
(SNP 9X9-9X FUEL LOADED IN CYCLE 7 WITH CHANNEL BOW)

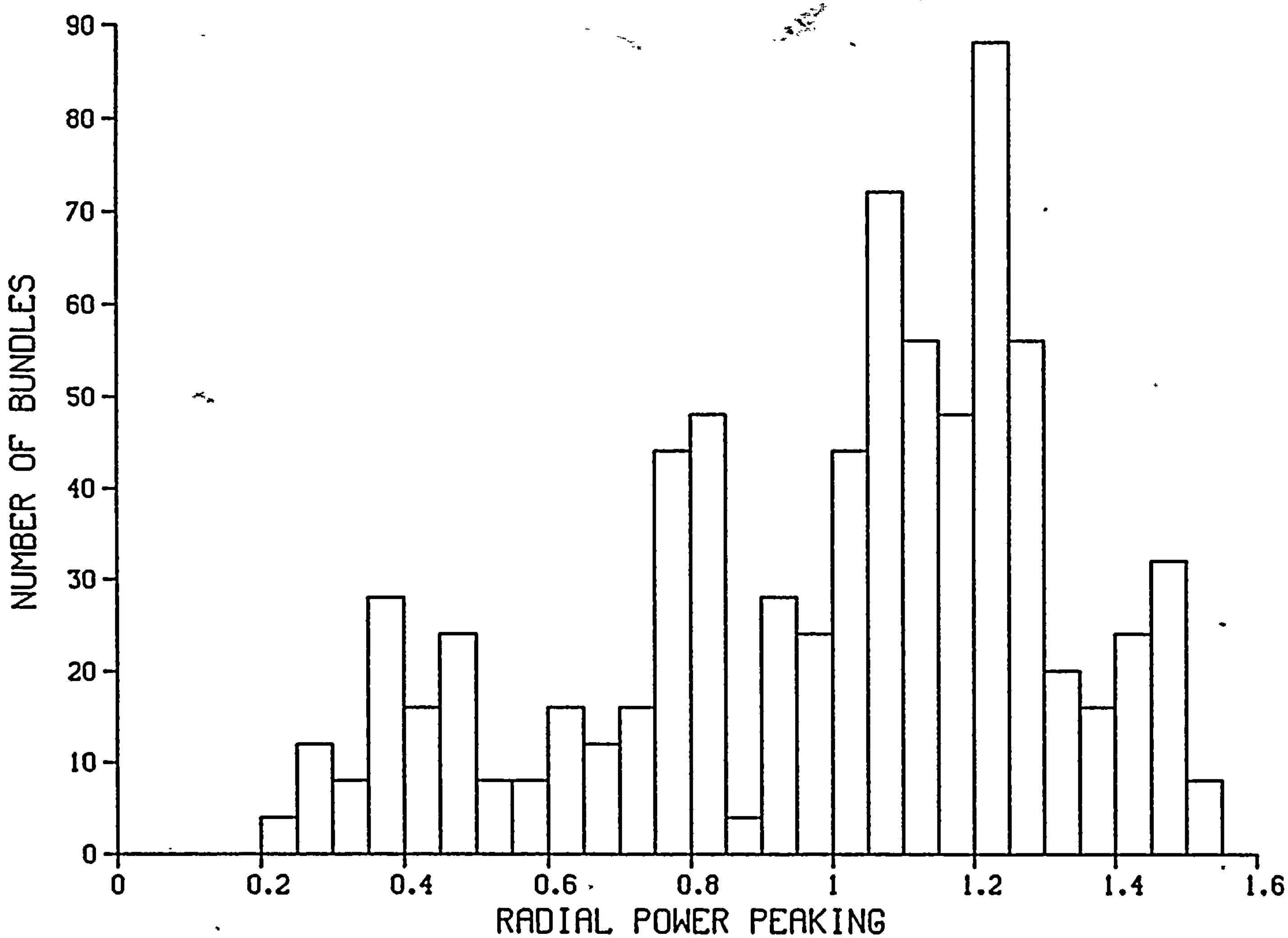

FIGURE A.3 RADIAL POWER HISTOGRAM FOR FULL CORE SAFETY LIMIT MODEL