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WNP-2

CYCLE 5 RELOAD SUMMARY REPORT

Washington Public Power Supply System
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WNP-2 CYCLE 5 RELOAD SUMMARY REPORT

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NOTICE

This report is derived in part through information provided to Washington Public Power Supply System (Supply System) by Advanced Nuclear Fuels Corporation. It is being submitted by the Supply System to the U.S. Nuclear Regulatory Commission in support of the WNP-2 Application For Technical Specifications Changes Relating to WNP-2 Cycle 5 operation. The information contained herein is true and correct to the best of the Supply System's knowledge, information, and belief.

WNP-2 CYCLE 5 RELOAD SUMMARY REPORT

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WNP-2 CYCLE 5 RELOAD SUMMARY REPORT

1.0 INTRODUCTION

The fourth reload of the Washington Public Power Supply System Plant No. 2 (WNP-2) will utilize Advanced Nuclear Fuels Corporation (ANF) 8x8 fuel plus four Lead Fuel Assembly (LFA) bundles utilizing a 9x9 pin array. The 8x8 fuel design of this reload batch is virtually identical to the fuel design of the previous reload batch. The 9x9 array LFA bundles represent an advanced design described in more detail in Section 6.0 of this report. This report summarizes the reload analyses performed by ANF in support of WNP-2 operation for Cycle 5. In addition, a description of the ANF reload is given along with a comparison of the characteristics of the Cycle 4 and Cycle 5 cores. A discussion of the proposed physics startup program is also included. The proposed license amendments (technical specification changes) are listed by title in this report for completeness.

The reload licensing submittal is composed of the WNP-2 Cycle 5 Reload Analysis Report (ANF-89-02) (Reference 1.0), the WNP-2 Cycle 5 Plant Transient Analysis Report (ANF-89-01) (Reference 2.0), the proposed changes to the WNP-2 Technical Specifications and this report. Where appropriate, this report summarizes analyses and makes reference to the above reports and other documents for detailed support. The WNP-2 Cycle 5 Reload Analysis Report (Reference 1.0) is intended to be used in conjunction with ANF Topical Report XN-NF-80-19(P)(A), Volume 4, Revision 1, Application of the ANF Methodology to BWR Reloads (Reference 3.0), which gives a detailed description of the methods and analyses utilized.

2.0 GENERAL DESCRIPTION OF RELOAD SCOPE

During the fourth refueling outage for WNP-2, the Supply System will replace 144 of the General Electric (GE) initial core fuel assemblies with ANF reload fuel. One hundred thirty-six (136) of the Cycle 5 reload fuel assemblies will be 8x8 current design reload assemblies which contain a bundle average enrichment of 2.62 weight percent U-235, four (4) of the Cycle 5 reload fuel assemblies will be 8x8 current design reload assemblies originally fabricated for use in Cycle 4 with a bundle average enrichment of 2.64 weight percent U-235 and four (4) of the Cycle 5 reload assemblies will be LFA's as described in Section 6.0 of this report. Two of the one hundred thirty six (136) 2.62 weight percent U-235 reload fuel assemblies have been manufactured with cladding having a composition controlled within a limited part of the ASTM specification for zircaloy-2. These assemblies are identical in all aspects to the others in this reload batch. This change in WNP-2 core loading required a partial re-analysis by ANF. The Loss of Coolant Accident (LOCA) and the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) analyses relevant to Cycle 5 operations are given in Reference 4.0 as these analyses were performed for all ANF fueled cores as a part of the Cycle 2 (initial reload) analysis. These analyses also bound operation of the

9x9 LFA fuel assemblies as discussed in Section 6.0. A MAPLHGR relationship for the 9x9 LFA's is included in the proposed technical specification change. The 9x9 MAPLHGR is based upon the above referenced MAPLHGR analysis for the 8x8 reload fuel and is different only to address the increased number of fuel rods in the 9x9 LFA bundles relative to the 8x8 reload fuel. A linear heat generation rate (LHGR) was developed for the 9x9 LFA fuel bundles and is included in the proposed Technical Specification changes. Relevant transient analyses and Minimum Critical Power Ratio (MCPR) analyses for the Cycle 5 loading are reported herein. These analyses identified the need for a revised reduced flow MCPR relationship for WNP-2, Cycle 5. This revised reduced flow MCPR relationship is included in the proposed Technical Specification changes. Analyses of normal reactor operation consisted of evaluation of the mechanical, thermal hydraulic, and nuclear design characteristics. Operation at extended core flow, single loop operation and final feedwater temperature reduction are also addressed.

A number of proposed changes to the WNP-2 Technical Specifications have resulted from the ANF design and safety analyses for the Cycle 5 core. A list of these Technical Specification changes is given in Table 2.1.

TABLE 2.1

PROPOSED TECHNICAL SPECIFICATION CHANGES

INDEX

B2.1.2	Thermal Power, High Pressure and High Flow*
3/4.1.3.4	Four Control Rod Group Scram Insertion Times**
3/4.2.1	Average Planar Linear Heat Generation Rate
3/4.2.3	Minimum Critical Power Ratio
3/4.2.4	Linear Heat Generation Rate
B3/4.2.1	Average Planar Linear Heat Generation Rate
5.3	Reactor Core

* Editorial change only to reflect change submitted and approved as Technical Specification Amendment No. 28.

**Included in this submittal for convenience. This change corrects a previous oversight in the Tech. Specs.

2.1 SUMMARY OF RESULTS

The limiting transient MCPR results for the analyses described in sections 8 and 9 of this document are summarized in columns 3 and 4 of Table 2.2.

WNP-2 will be entering its fifth cycle of operation and is approaching an equilibrium cycle. Analysis results between Cycles 3, 4 and 5 have shown little change. For Cycle 4 operation, WNP-2 chose to add a small CPR penalty for margin to envelope future anticipated analysis results. The resulting CPR limits, including these self-imposed penalties, are summarized in columns 5 and 6 of Table 2.2. The CPR limits listed for ANF fuel in column 4 are applicable to both the ANF 8x8 and ANF 9x9 fuel designs included in this reload. The values listed in columns 3 and 4 of Table 2.2 form the basis for the requested changes to Tech. Spec. Table 3.2.3-1 for WNP-2 for Cycle 5 operations.

TABLE 2.2

MCPR OPERATING LIMITS

1.	2.	3.	4.	5.	6.
		Cycle-5 Analysis -- Results		Cycle-4 WNP-2 Tech Spec Values	
Cycle Explosure	Equipment Status	GE Fuel	ANF Fuel	GE Fuel	ANF Fuel
0-3750 MWD/MT	*	1.23	1.23	1.40	1.28
3750 - EOC	Normal SCRAM times	1.34	1.31	1.40	1.31
3750 - EOC	TS SCRAM times	1.41	1.37	1.50	1.38
3750 - EOC	RPT inop Normal SCRAM times	1.41	1.37	1.50	1.37
3750 - EOC	RPT inop TS SCRAM times	1.47	1.41	1.55	1.43
0 - EOC	Single Loop Operation	1.35	1.35	1.40	1.37

*In this portion of the fuel cycle, operation with the given MCPR operating limits is allowed for both normal and Tech. Spec. scram times and for both RPT operable and inoperable.

3.0 WNP-2 CYCLE 4 OPERATING HISTORY

WNP-2, a 3323 mwt BWR 5, began Cycle 4 operation on June 23, 1988*. The end of Cycle 4 operation is expected to be April 14, 1989. During Cycle 4, the plant was base loaded at or near 100 percent power for all of the cycle.

Figure 3:1 gives a power history of Cycle 4 through January 1989, for WNP-2. The Cycle 4 operating highlights and control rod sequence exchange schedule are found in Table 3.1.

* Significant thermal operation began on this date. Significant electrical generation commenced on June 28, 1988.

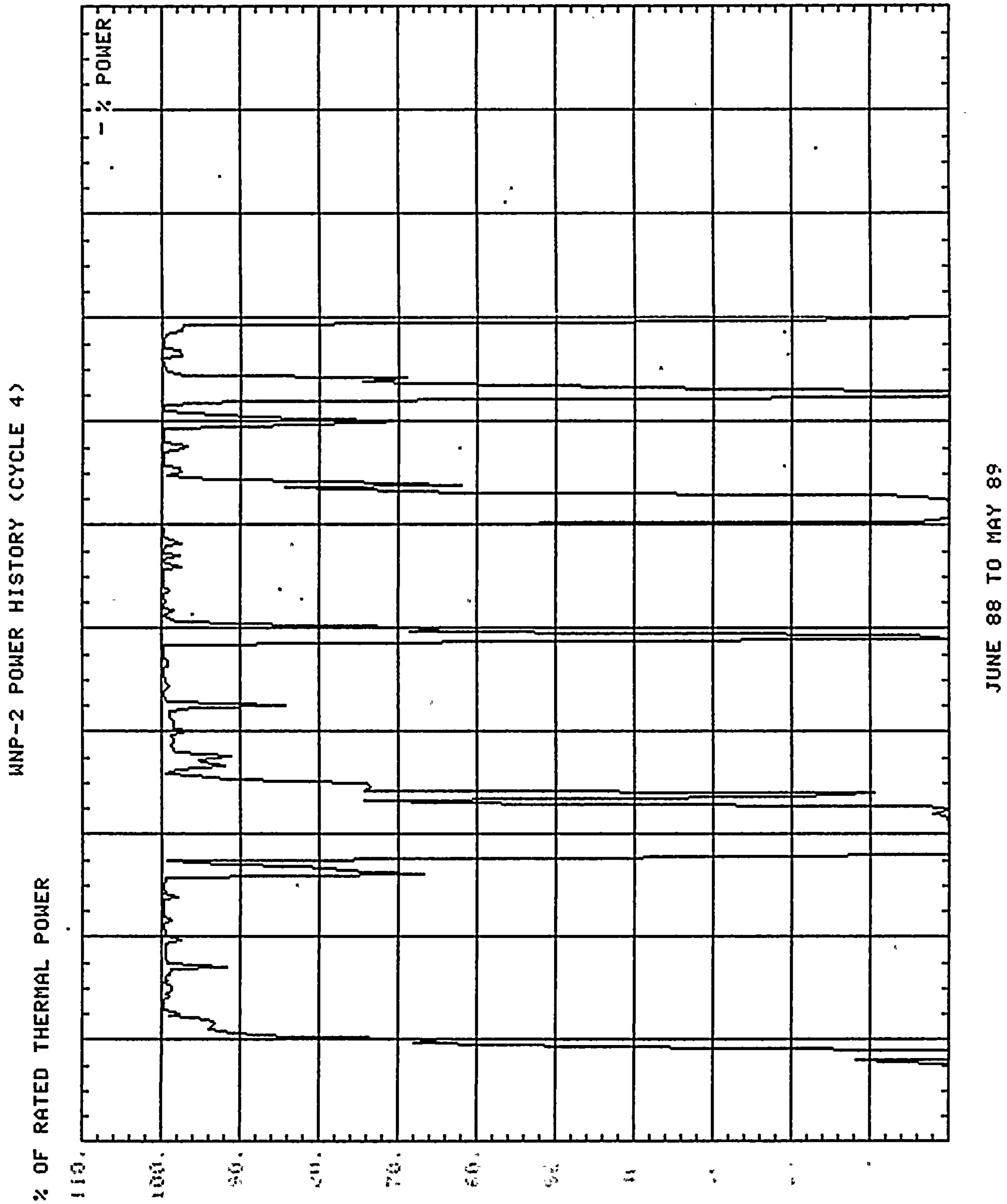


Figure 3.1
Power History for WNP-2 for Cycle 4

TABLE 3.1

WNP-2 CYCLE 4 OPERATING HIGHLIGHTS

Began Fuel Loading	May 10, 1988
Began Electric Power Production	June 28, 1988
Projected End of Cycle Date	April 14, 1989
End of Cycle Core Average Exposure (Design)(mwd/mtm)	16,870
Number of Fresh Assemblies	152
Gross Generation (FPD) (projected)	237

Control Rod Sequence Exchange Schedule

	Sequence	
Date	From	To
August 19, 1988	A2	B2
October 27, 1988	B2	A1
December 30, 1989	A1	B1
February 22, 1989*	B1	A2

Outages

August 24 through September 8, 1988

October 27 through October 29, 1988

December 1 through December 8, 1988

January 6 through January 9, 1989

January 30 through February 2, 1989

*Estimated date; also, may alternately choose to go to A-1 sequence.

4.0 RELOAD CORE DESCRIPTION

The WNP-2 core consists of 764 fuel assemblies. For the Cycle 5 reload, the core will consist of 144 ANF fresh assemblies, 152 ANF 8x8C assemblies loaded for Cycle 4, 148 ANF 8x8C assemblies loaded for Cycle 3, 128 ENC 8x8C fuel assemblies loaded for Cycle 2 and 192 GE 8x8RP assemblies remaining from the initial core. The 144 ANF fresh assemblies consist of four reload 8x8C assemblies originally manufactured for loading in Cycle 4 (ANF-3), 136 reload 8x8C assemblies manufactured for loading in Cycle 5 (ANF-4) and four 9x9 LFA reload assemblies. The four ANF-3 assemblies have a bundle average enrichment of 2.64 weight percent U-235 and the 136 reload ANF-4 fuel assemblies have a bundle average enrichment of 2.62 weight percent U-235. The four ANF-3 8x8C assemblies have five fuel rods containing gadolinium oxide (Gd_2O_3) at a loading of 2.0 weight percent as a neutron poison. The 136 ANF-4 8x8C assemblies have six fuel rods containing gadolinium oxide (Gd_2O_3) at a loading of 2.0 weight percent. Minor differences, primarily in end plug design, exist between the two assembly 8x8C designs. The four 9x9 LFA reload assemblies are described in detail in Section 6.0. The assembly designs are interchangeable with regard to all of the analyses reported here. Table 4.1 lists the assembly type, quantity, and initial enrichment for the assemblies which will make up the Cycle 5 core.

TABLE 4.1

WNP-2 CYCLE 5 CORE

Number of Assemblies	Type	Enrichment
140*	ANF 8x8C	2.62/2.64 w/o U-235
4	ANF 9x9	2.53/2.59 w/o U-235
152**	ANF 8x8C	2.64/2.72 w/o U-235
148***	ANF 8x8C	2.72 w/o U-235
128****	ENC 8x8C	2.72 w/o U-235
136	GE P8x8R	2.19 w/o U-235
56	GE P8x8R	1.76 w/o U-235

The 144 exposed GE P8x8R assemblies to be discharged are all high enriched (2.19 w/o U-235) assemblies.

*Four (4) of these assemblies were fabricated for reload in Cycle 4 (ANF-3) and have an enrichment of 2.64 weight percent U-235 and one hundred thirty-six (136) of these assemblies were fabricated for reload in Cycle 5 (ANF-4) and have an enrichment of 2.62 weight percent. Two (2) of these ANF-4 assemblies contain characterized fuel pins.

**Twenty-four (24) of these assemblies were originally fabricated for reload in Cycle 3 (XN-2) and have an enrichment of 2.72 weight percent U-235 and one hundred twenty-eight (128) of these assemblies were fabricated for reload in Cycle 4 (ANF-3) and have an enrichment of 2.64 weight percent U-235. Two of these ANF-3 assemblies are Lead Test Assemblies (LTA) of standard ANF 8x8C design.

***Thirty-six (36) of these assemblies were originally fabricated for reload in Cycle 2 (XN-1) and one hundred twelve (112) of these were fabricated for reload in Cycle 3 (XN-2). They are effectively identical.

****Two of these assemblies are Lead Test Assemblies (LTA) of standard ANF 8x8C design.

5.0 FUEL MECHANICAL DESIGN (8x8C FUEL)

The mechanical design of the 8x8C Cycle 5 ANF reload fuel for WNP-2 is described briefly in Reference 5.0 and more completely in References 6.0, 7.0 and 8.0. This fuel is essentially identical to the 8x8C Cycle 2 ENC fuel described in Reference 4.0. The fuel assembly design uses 62 fuel rods and two centrally located water rods, one of which functions as a spacer capture rod. Seven spacers maintain fuel rod pitch. The design uses a quick-removable upper tie plate design to facilitate fuel inspection and bundle reconstitution of irradiated assemblies. The fuel rods utilize Zircaloy-2 cladding, 35 mils thick. The fuel rods are pressurized, and contain either UO_2 - Gd_2O_3 or UO_2 with a nominal density of 94.5 percent of theoretical density (TD), and an 8.5 mil nominal diametrical pellet to clad gap for the enriched pellets. Natural uranium is loaded in the top and bottom six inches of each fuel rod for greater neutron economy. The enriched pellets have a slightly larger diameter than the natural pellets.

The fuel mechanical design analysis performed on the ANF 8x8C reload fuel evaluated the following items (Reference 8.0):

- Cladding steady state strain and stress.
- Transient strain and stress.
- Cladding fatigue damage.
- Creep collapse.
- Corrosion.
- Hydrogen absorption.
- Fuel rod internal pressure.
- Differential fuel rod growth.
- Creep bow.
- Grid space design.

The analyses presented in Reference 8.0 justify irradiation to a 35,000 MWD/MT peak assembly burnup in WNP-2.

Some major results of these analyses are:

- The maximum end-of-life (EOL) steady state cladding strain is well below the 1 percent design limit.
- Cladding steady state stresses are calculated below the material strength limits.

- The transient strain does not exceed 1.0 percent.

- The cladding fatigue usage factor is within the 0.67 percent design limit.

- The cladding diameter reduction due to uniform creepdown, plus creep ovality at maximum densification, is less than the minimum initial gap. Compliance with this criteria prevents the formation of fuel column gaps and the possibility of creep collapse.

- The maximum level of the corrosion layer was calculated to be well within the design limit.

- The maximum concentration of hydrogen was calculated to be well within the design limit.

- Evaluations of the fuel assembly growth and differential fuel rod growth show that the fuel assembly design provides adequate clearance.

- The plenum spring complies with design limits.

- The spacer spring meets all design requirements.

- The maximum fuel rod internal rod pressure remains below ANF's criteria limit.

- The fuel centerline temperature remains below the melting point.

The structural response of the 8x8C ANF-4 reload fuel is the same as the structural response of the 8x8C ANF-3 fuel, the 8x8C XN-2 ANF fuel, the 8x8C XN-1 fuel and the P8x8R GE fuel which also reside in the WNP-2 Cycle 5 core. As a part of Cycle 5 operation, some of the 8x8C Cycle 4 ANF reload fuel assemblies may be channeled with new 100 mil channels fabricated by ASEA Brown Boveri (ABB) and/or Cartech as has been the practice in the past. These channels are equivalent to the initial core channels. The remainder of the reload fuel bundles will be channeled with channels which have been previously discharged from WNP-2. Prior to reuse, these channels are measured with the WNP-2 channel measuring machine and qualified for reuse based upon a predetermined criteria. Therefore, the seismic LOCA structural response evaluation performed in support of the initial core remains applicable and continues to provide assurance that control blade insertions will not be inhibited following occurrence of the design basis seismic LOCA event.

A LHGR limit is placed on ANF 8x8C Cycle 5 reload fuel assemblies for monitoring for the reasons given previously in Reference 4.0, Page 10, for ENC 8x8C Cycle 2 fuel.

6.0 LEAD FUEL ASSEMBLY PROGRAM

The lead fuel assembly program for Cycle 5 of WNP-2 consists of four fuel assemblies all of which utilize a 9x9 fuel pin array with an interior water channel which displaces the central 3x3 fuel pin configuration of the assembly and which functions as a spacer/capture device. Two of the LFA assemblies are designated 9x9-IX and two of the assemblies are designated 9x9-9X. The 9x9-IX bundles incorporate the use of a zirconium liner within the cladding on all fuel pins except for those pins containing Gd_2O_3. These fuel pins are clad with beta quenched zircaloy cladding as is used in the rest of the Gd_2O_3 fuel pins in this reload. The 9x9-9X bundles do not utilize a pure zirconium liner and thus are made with standard zircaloy cladding except for the Gd_2O_3 fuel pins which use beta quenched clad.

Table 6.1 gives a comparison of key parameters for the ANF-4 8x8C bundle, the 9x9-IX bundle and the 9x9-9X bundle. The lead fuel assemblies are placed in non-limiting core locations within the reload core.

TABLE 6.1

PARAMETER	Fuel Assembly Design Parameters 8x8C	9x9-IX	9x9-9X
Number of Rods, Total	64	72	72
Fuel Rod Pitch, Inches	.641	.569	.569
Fuel Assembly Loading, kg UO^2	199.7	201.0	190.6
Fuel Assembly Loading, KgU	176.0	176.8	167.7
Fuel Pellet			
Material	UO_2	UO_2	UO_2
Density g/cc	10.36	10.55 *	10.36
Percent of T.D.	94.5	96.26 *	94.5
Diameter			
Enriched	.4055	.374*	.3665
Natural	.4045	.374*	.3665
Dish Volume (Percent of pellet Volume)			
Enriched UO_2	1.50	1.00	1.00
Enriched UO_2 - GD_2O_3	1.00	1.00	1.00
Natural	0.00	1.00	1.00
Fuel Rod (6" Natural Enriched Material on each end)			
Fuel Length	150"	150"	150"
Cladding Material	Zr-2**	Zr-2/Zr Liner**	Zr-2**
Clad I.D., Inches	.414	.3807 *	.373
Clad O.D., Inches	.484	.433	.433
Enriched Length	138"	138"	138"
Fuel Rod Inventory			
Low-Low Enrichment	1 (1.5 w/o U-235)		
Low Enrichment	5 (2.0 w/o U-235)	4 (1.92 w/o U-235)	4 (1.92 w/o U-235)
Medium Low Enrichment	9 (2.5 w/o U-235)		
Medium Low Enrichment with Gd_2O_3	6 (2.5 w/o U-235 +2.0 w/o Gd_2O_3)	5 (2.51 w/o U-235 +1.8 w/o Gd_2O_3) 1 (2.51 w/o U-235 +4.5 w/o Gd_2O_3)	5 (2.51 w/o 6.235 + 1.8% Gd_2O_3) 1 (2.51 w/o 2.235 + 4.5 w/o Gd_2O_3)
Medium Enrichment	21 (2.64 w/o U-235	12 (2.51 w/o U-235)	12 (2.51 2/o U-235)
High Enrichment	20 (3.43 w/o U-235)	50 (2.82 w/o U-235)	50 (2.90 w/o &-235
Inert Water Rod	2	Large Central Can	Large Central Can
Unvoid Water Area (Inches)	.368	2.447	2.447
Maximum Width (Inches)	5.251	5.252	5.252

*Gd_2O_3 Rods use 9x9-9X design parameters
**Gd_2O_3 Rods use Zr-2 (Beta Heat Treated)

Analyses have been performed consistent with ANF methodology (references 3.0, 10.0 and 12.0) to establish a licensing basis for the two ANF 9x9-IX and the two ANF 9x9-9X LFA's in the WNP-2 Cycle 5 core. The analyses demonstrate the applicability of the WNP-2 Cycle 5 operating limits to these four LFA's unless stated otherwise.

The insertion of four ANF 9x9 LFA's in the Cycle 5 core will have negligible effects upon core wide transient performance. However, some 9x9 LFA specific analyses have been performed to assure that the Cycle 5 operating limits are also applicable to the LFA's. Fuel specific LHGR and MAPLHGR limits have been developed for these LFA's.

The dynamic response of the LFA is expected to be almost identical to that of the 8x8 already in the core. This is due to the fact that the fuel assembly stiffness is provided by the assembly channel, which is the same in both designs.

The 9x9 LFA's are hydraulically compatible with the co-resident ANF 8x8 fuel assemblies based on a comparison of fuel component hydraulic resistances.

Steady state thermal hydraulic analysis has shown that even though the ANF 9x9 LFA design has a somewhat smaller flow area than the ANF 8x8 design, no reduction in thermal margin is experienced in the Cycle 5 core. This is due to the increased critical power performance of the ANF 9x9 design relative to the ANF 8x8 design at WNP-2 Cycle 5 conditions.

The average enrichment and enrichment distribution for the 9x9-IX and 9x9-9X fuel assemblies have been selected to match, as closely as possible, the neutronic performance of the four 8x8 ANF-3 2.64 w/o U-235 reload assemblies included in the Cycle 5 reload. Based on this selection, the neutronic characteristics of the 9x9 LFA's are expected to match to the characteristics of the ANF 8x8 fuel included in the Cycle 5 reload.

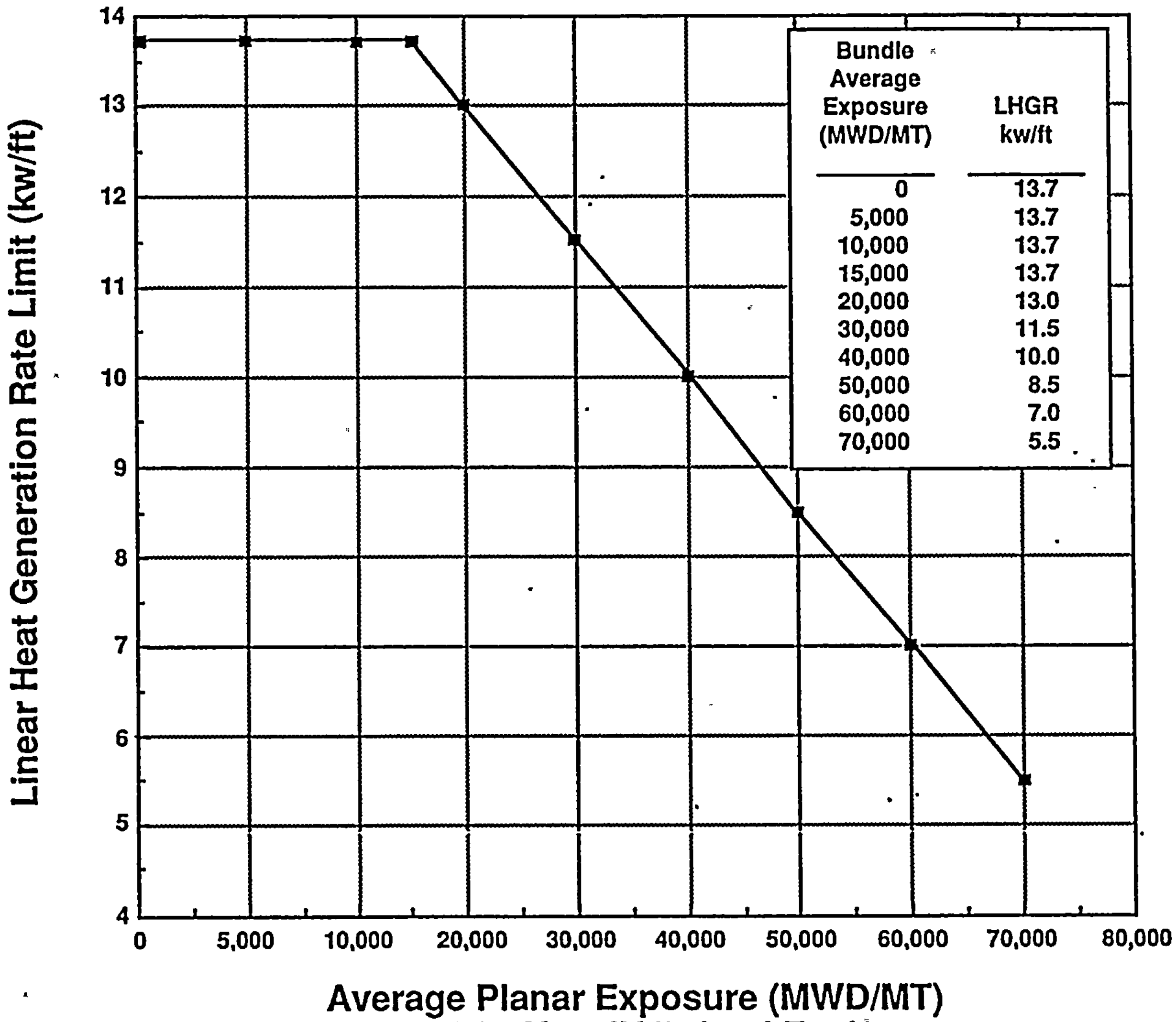
Analyses of the WNP-2 Cycle 5 limiting transients have been performed for ANF 8x8, ANF 9x9 LFA and GE P8x8R fuel. It has been shown that using the XN-3 ANF Critical Heat Flux (CHF) correlation, the bundle power required to produce transition boiling in an ANF 9x9 LFA is higher than that for an ANF 8x8 bundle. The Cycle 5 Safety Limit Analysis considered the LFA's such that the MCPR safety limit of 1.06 is also applicable to the LFA's. Therefore, the ANF 9x9 LFA's can be monitored to the ANF 8x8 MCPR fuel limits.

Since heatup is primarily a planar and not an axial phenomena, the appropriate bundle power limit derived from a LOCA analysis is the peak bundle planar power. The ANF 9x9 LFA's have better cooling during LOCA conditions relative to an ANF 8x8 fuel assembly due to the lower stored energy in the fuel rods, a greater surface area provided by the larger number of fuel rods, and more inert surface from the central water channel. Thus, a

LOCA analysis for the ANF 9x9 LFA's would yield lower Peak Cladding Temperatures (PCT's) and metal-water reaction rates than an ANF 8x8 assembly at the same bundle peak planar power. The MAPLHGR limits for the ANF 9x9 LFA's restrict the peak bundle planar power to that analyzed for the ANF 8x8 fuel and assure that the USNRC 10CFR50.46 criteria are met for the ANF 9x9 LFA's in Cycle 5.

The fuel loading error was analyzed for the ANF 9x9 LFA's. Results show that if the loading error went undetected, the offsite consequences would remain well within the guidelines specified in 10CFR Part 100.

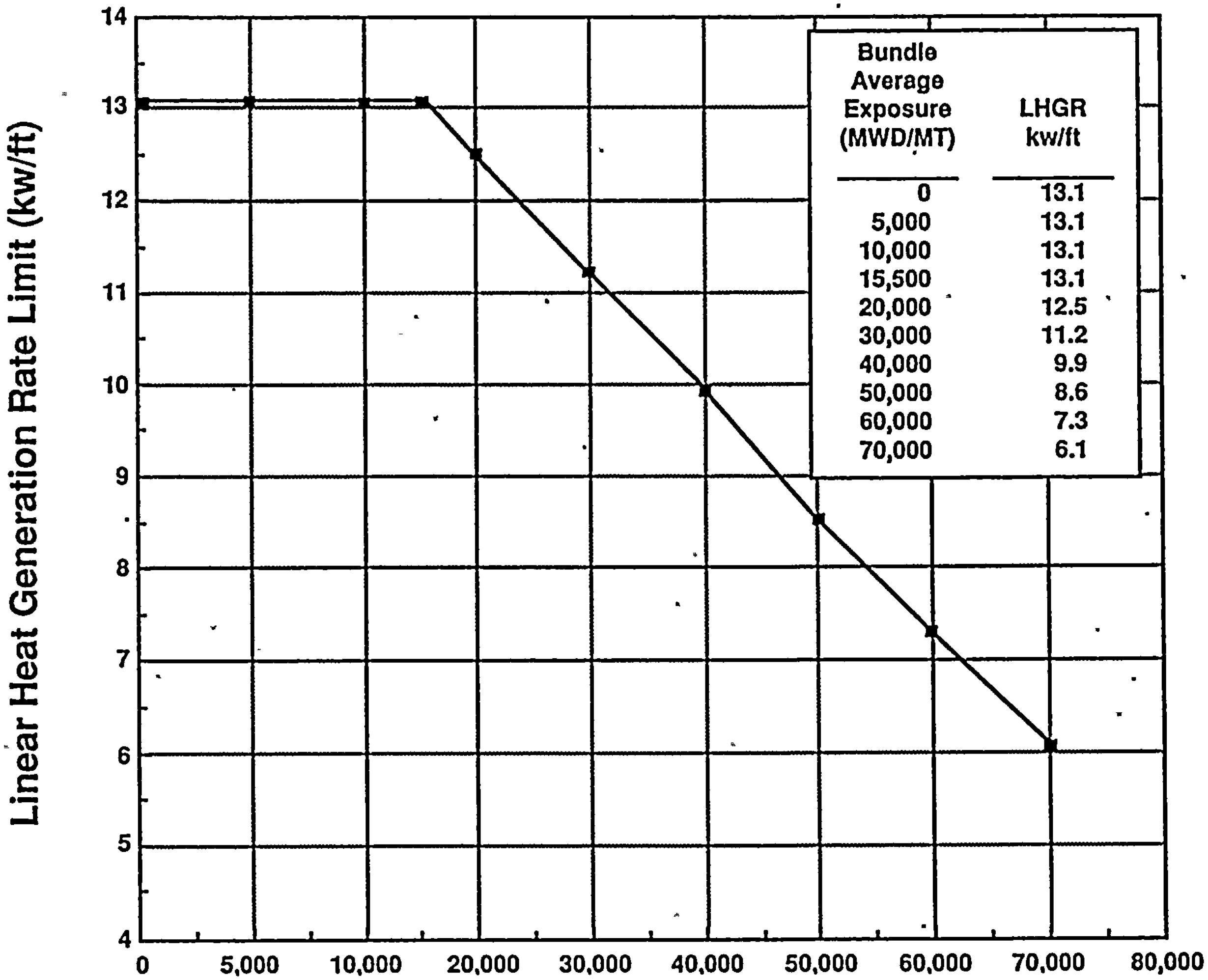
All operational limits used for ANF 8x8 fuel are applicable to the ANF 9x9 LFA's except for fuel type specific MAPLHGR limits and the 9x9-IX and 9x9-9X LHGR limits. The LHGR limits for the 9x9-IX and 9x9-9X are shown in Figures 6.1 and 6.2 respectively, and the MAPLHGR limits for the LFA's are shown in Figure 6.3.



ANF 9 X 9 - IX Reload Fuel
Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure

Figure 6.1

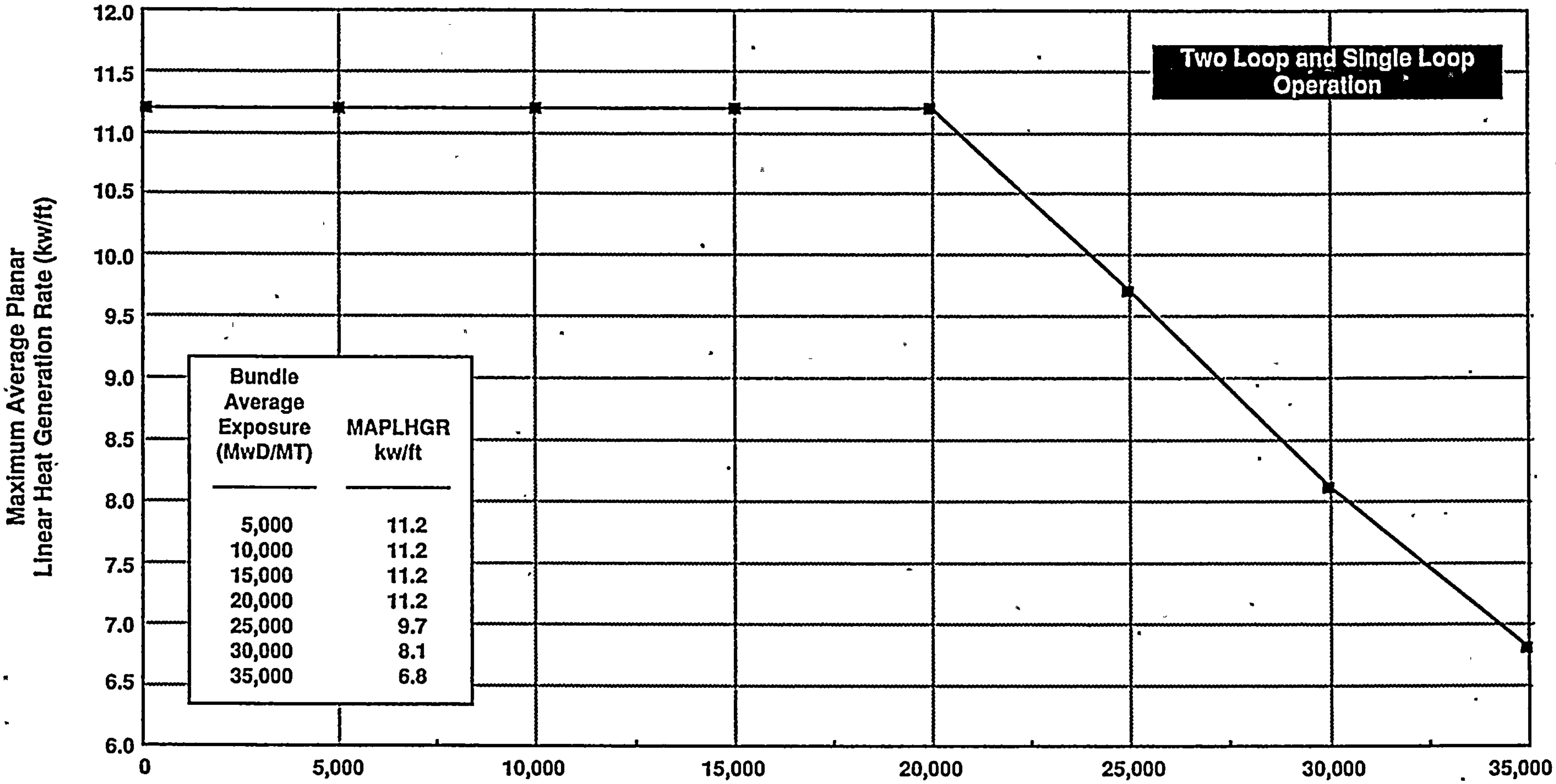
890040.3



Average Planar Exposure (MWD/MT)
ANF 9 X 9 - 9X Reload Fuel
Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure

Figure 6.2

890040.4



Bundle Average Exposure (MWD/MT)
ANF 9 X 9 - IX AND 9 X 9 - 9X Reload Fuel
Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Bundle Average Exposure

Figure 6.3

890040.2a

7.0 THERMAL HYDRAULIC DESIGN

The goal of the thermal hydraulic design analysis is to demonstrate that the ANF reload fuel meets and/or exceeds the primary thermal hydraulic design criteria. Principal design criteria considered in the thermal hydraulic analysis are found in XN-NF-80-19(A), Volume 4, Revision 1 (Reference 3.0).

Analyses performed to demonstrate that these criteria are met include:

- Hydraulic compatability.
- Fuel cladding integrity safety limit.
- Fuel centerline temperature.
- Bypass flow characteristics.
- Thermal hydraulic stability.

These analyses are discussed in this section for 8x8 fuel. Specific thermal hydraulic design considerations for the 9x9 fuel are discussed in Section 6.0.

7.1 Hydraulic Compatability

The hydraulic flow resistances for the ANF reload fuel and the GE 8x8 fuel have been determined in single phase flow tests of full scale assemblies. XN-NF-80-19(A), Volume 4, Revision 1 (Reference 3.0), reports the resistances measured and evaluates the effects on thermal margin of mixed ANF and GE 8x8 cores. The close geometrical similarity between the fuel designs and their measured performance characteristics demonstrate that the fuel designs are sufficiently compatible for co-residence in WNP-2. Hydraulic compatability of the 9x9 reload fuel is discussed in Section 6.0.

7.2 Fuel Cladding Integrity Safety Limit

The MCPR fuel cladding integrity safety limit for Cycle 5 is 1.06 which is equal to the Cycle 1, Cycle 2, Cycle 3 and Cycle 4 MCPR safety limit. The methodology used in the MCPR safety limit calculations is found in XN-NF-80-19(A), Volume 4, Revision 1 (Reference 3.0). The WNP-2 Cycle 5 MCPR safety limit analysis methodology and input parameters are described in ANF-89-01, Cycle 5 Plant Transient Report (Reference 2.0).

7.3 Fuel Centerline Temperature

The LHGR curve in Figure 3.4 of Reference 8.0 for 8x8 fuel and the LHGR curves in Figures 6.1 and 6.2 for 9x9 fuel are everywhere greater than 120 percent of the LHGR limit curve in Reference 8.0. Therefore, the fuel centerline temperature is protected for 120 percent over power and fuel centerline melt is protected for all fuel exposures within the bounds of the referenced LHGR curve.

7.4 Bypass Flow Characteristics

Core bypass flow was computed using the methodology of XN-NF-524(A) (Reference 9.0). The bypass flow for the WNP-2 Cycle 5 is 10.7 percent of the total core flow which is similar to the Cycle 1 value of 11.8 percent and to the Cycle 4 value of 11.5 percent. The computed bypass flow will have no adverse impact on reactor operation.

7.5 Thermal Hydraulic Stability

The WNP-2 Technical Specifications included surveillance requirements for detecting and suppressing power oscillations. In addition, the ANF COTRAN code (Reference 10.0) was used to specifically determine that the worst case value of decay ratio is less than 0.60 in the area of the power flow map bounded by the APRM rod block line at 45 percent rated flow. The worst case decay ratio is no greater than 0.9 in the area of allowable low flow operation (detect and suppress region). The bounding power flow points in the detect and suppress region are the APRM rod block line at 27.6 percent core flow (47 percent power - minimum allowable two pump flow) and the APRM rod block line at 23.8 percent core flow (42 percent power-natural circulation) (Reference 1.0).

The 9x9 LFA's were included in the core-wide stability analysis. Local instability tests were performed on 9x9 loads in a BWR-3. No detectable difference was noted in stability performance relative to the co-resident 8x8 fuel (Reference 1.0).

8.0 NUCLEAR DESIGN

The neutronic methods for the design and analysis of the WNP-2 Cycle 5 reload are described in Reference 10.0. These methods have been reviewed and approved by the U.S. Nuclear Regulatory Commission for generic application to BWR reloads.

8.1 Fuel Bundle Nuclear Design

The Cycle 5 8x8 ANF reload bundles (labeled ANF-4) are similar to the ANF-3 ANF reload bundles (used in the WNP-2 Cycle 4 reload) in nuclear design in all major parameters except for fuel enrichment. Major nuclear design characteristics for the ANF reload fuel assemblies (ANF-4) are:

- The 8x8C fuel assembly contains 62 fuel rods and two water rods. One of the water rods also acts as a spacer capture rod. The 9x9 LFA fuel assembly contains 72 fuel rods and one central water channel.

- The ANF-4 8x8C fuel assembly average enrichment is 2.62 w/o U-235. The top and bottom six inches of the fuel rods contain natural uranium. The central 138 inch portion of the fuel rods has an average enrichment of 2.79 w/o U-235. The 9x9 LFA fuel assembly average enrichments are 2.53 and 2.59 w/o U-235 with six inches of natural uranium at the top and bottom.

- Five enrichment levels are utilized in the ANF-4 8x8C fuel assembly to produce a local power distribution which results in a balanced design for Minimum Critical Power Ratio (MCPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits. The 9x9 LFA fuel designs utilize three enrichment levels to accomplish the same goal.

- Each ANF-4 8x8C fuel assembly contains six fuel rods with 2.0 w/o GD_2O_3 blended with 2.50 w/o U-235 enriched UO_2 to reduce initial assembly reactivity. The 9x9 LFA fuel assemblies contains five 1.8% w/o Gd_2O_3 pins and one 4.5 w/o Gd_2O_3 pin all enriched to 2.51 w/o U-235 for the same purpose.

The enrichment distribution of the ANF reload designs was selected on the basis of maintaining a balance between the local power peaking factors, assembly reactivity, MAPLHGR, and MCPR. For the central enriched region of the AN-4 8x8 assembly, one rod is enriched to 1.5 w/o U-235, five rods to 2.0 w/o U-235, nine rods to 2.50 w/o U-235, 21 rods to 2.64 w/o U-235, 20 rods to 3.43 w/o U-235, and six rods to 2.50 w/o U-235 plus 2.00 w/o GD_2O_3. The fuel bundle design features of the 9x9 LFA designs can be found in Table 6.1. In this table, a comparison is also made to the design characteristics of the 8x8 ANF-4 design.

8.2 Core Nuclear Design

The core exposure for the end of Cycle 4 (EOC4), the core exposure for the beginning of Cycle 5 (BOC5), and the core exposure for the end of Cycle 5 (EOC5) were calculated with the XTGBWR Code (Reference 10.0). In addition, BOC core reactivity characteristics for the cold core were calculated along with the standby liquid control system reactivity. The results of these analyses are summarized in Table 8.1.

Table 8.1

CORE NUCLEAR DESIGN

Core Exposures at EOC4 (mwd/mtm)	16,700
Core Exposures at BOC5 (mwd/mtm)	12,300
Core Exposures at EOC5 (mwd/mtm)	18,100
BOC Cold K_{eff}, all rods out	1.1133
BOC Cold K_{eff}, strongest rod out	0.9868
Reactivity Defect/R-Value, percent % $\Delta K/K$	0.0
Standby Liquid Control System (SBLC) Reactivity, 660 PPM Boron, K_{eff}	0.9633

8.3 Comparison of Major Core Parameters

Some of the major core parameters for WNP-2 Cycle 4 and Cycle 5 are listed in Table 8.2.

Table 8.2

COMPARISON OF MAJOR CORE PARAMETERS

Parameter	Cycle 4	Cycle 5
MCPR Limit* (0 mwd/mtm)	1.23	1.23
Doppler Defect (% $\Delta K/K/T$)	-9.5×10^{-6}	-10.0×10^{-6}
Cycle Length** (Design; FPD)	227	227
Core Average Exposure (BOC; mwd/mtm)	11,200	12,300
Core Average Exposure (EOC; mwd/mtm)	16,700	18,100

* Based on CRWE, 106% RBM setpoint
**All rods out; full power, 106% flow

The differences between the Cycle 4 core and the Cycle 5 core are found in the core loading pattern. The Cycle 4 core consisted of a scatter load of 152 ANF 8x8C unirradiated reload assemblies, 148 once irradiated ANF 8x8C reload assemblies, 128 twice irradiated ANF 8x8C reload assemblies and 336 GE P8x8R initial core assemblies. The Cycle 5 core will consist of a scatter load of 144 ANF 8x8C unirradiated reload assemblies, 152 once irradiated ANF 8x8C reload assemblies, 148 twice irradiated 8x8C reload assemblies, 128 thrice irradiated 8x8C reload assemblies and 192 P8x8R assemblies fabricated by General Electric (GE) left over from the initial core load.

9.0 ANTICIPATED OPERATIONAL OCCURRENCES

ANF considers eight categories of potential system core wide transient occurrences for jet pump BWRs (Reference 11.0) and has provided analysis results for the three most limiting transients for WNP-2 Cycle 5 to determine the Cycle 5 thermal margins. The three transients determined to be most limiting for Cycle 5 are:

- o Load Rejection No Bypass (LRNB).
- o Feedwater Controller Failure (FWCF).
- o Loss of Feedwater Heating (LOFH).

ANF's methodology for developing thermal limits is found in Reference 12.0. The discussion in Reference 11.0 demonstrates that the other plant transient events are inherently nonlimiting or clearly bounded by the above events.

Two local events, Control Rod Withdrawal Error (CRWE) and Fuel Loading Error (FLE) were analyzed with the methodology described in Reference 10.0. The CRWE was demonstrated to be bounding for certain parts of the fuel cycle.

The analysis reported here is applicable to both the 8x8C and 9x9 LFA fuel included in the Cycle 5 reload (References 1.0 and 2.0). The results of the core-wide and local transient analyses are provided in the WNP-2 Cycle 5 Reload Analysis Report (Reference 1.0) and in the WNP-2 Cycle 5 Transient Analysis Report (Reference 2.0). The CRWE was evaluated and found to be most limiting up to EOC-2000 mwd/mtm at 106 percent of rated core flow, resulting in a change in CPR of 0.17 for the ANF fuel and 0.17 for the GE fuel at the 106 percent rod block monitor (RBM) trip setpoint. When combined with the 1.06 safety limit, this transient (CRWE) requires a MCPR operating limit of 1.23 for the ANF fuel and 1.23 for the GE fuel in Cycle 5 in the range from BOC to EOC-2000 mwd/mtm. The ANF reload safety analyses were performed using control rod insertion times based on plant data. For operation in the range of EOC-2000 mwd/mtm to EOC up to 106 percent core flow with these normal scram times, the LRNB transient was determined to be the limiting transient and the MCPR limit for ANF fuel is 1.31 and for GE fuel is 1.34 for this portion of the fuel cycle. In the event that plant surveillance demonstrates that these scram insertion times are exceeded, the plant thermal margins default to values which correspond to the Technical Specification insertion times (3.1.3.4, P 3/4.1.8) for this portion of the fuel cycle (EOC-2000 mwd/mtm to EOC). For operation at and beyond EOC-2000 with core flow up to 106 percent and these technical specification scram times, the limiting transient is the LRNB transient and the MCPR operating limit within EOC-2000 mwd/mtm to EOC is 1.37 for ANF fuel and 1.41 for GE fuel for Cycle 5 operation. If the Recirculation Pump Trip (RPT) should become inoperable for any reason and assuming normal scram speeds, and operation up to 106 percent core flow in this exposure range, the limiting transient is then the LRNB transient and the MCPR operating limit is 1.37 for ANF fuel and 1.41 for GE fuel. Finally, if the RPT becomes inoperable within EOC-2000 mwd/mtm to EOC and the plant defaults to the technical specification scram times, the LRNB transient at 106 percent flow is bounding and the MCPR operating limit is 1.41 for ANF fuel and 1.47 for GE fuel.

Additional analyses were performed to determine the MCPR operating limit with a 107 percent and 108 percent RBM setpoint for the CRWE event. The resulting changes in CPR are 0.19 for ANF fuel and, 0.19 for GE fuel at 107 percent, and 0.21 for ANF fuel, and 0.21 for GE fuel at a 108 percent rod block setting. Therefore, operation with a 108 percent RBM setting would require a MCPR limit of 1.27 for ANF and 1.27 for the GE fuel.

9.1 Core Wide Transients

The plant transient model used to evaluate the pressurization transients, the LRNB and FWCF events, consists of the ANF COTRANSA (Reference 11.0) and XCOBRA-T (Reference 13.0) codes. This axial one dimensional model predicted reactor power shifts toward the core middle and top as pressurization occurred. This phenomenom was accounted for explicitly in determining thermal margin changes in the transient. All pressurization transients were analyzed on a bounding basis using COTRANSA in conjunction with the XCOBRA-T hot channel model. The LRNB event was found to be the most limiting core wide event at 106 percent core flow at EOC utilizing normal scram times. For technical specifications scram times, the LRNB event was found to be the most limiting core wide event at 106 percent core flow and EOC. With RPT inoperable and normal scram times, the LRNB event was found to be the most limiting core wide event at 106 percent core flow and EOC. With RPT inoperable and technical specification scram times, the LRNB was found to be the most limiting transient at 106 percent core flow and EOC. All core wide transients were analyzed using bounding values as input.

The Loss of Feedwater Heating (LOFH) events were evaluated with the ANF core simulator model XTGBWR (Reference 10.0) by representing the reactor in equilibrium before and after the event. Actual and projected operating statepoints were used as initial conditions. Final conditions were determined by reducing the feedwater temperature by 100°F and increasing core power such that the calculated eigenvalue remained unchanged.

Based on a bounding value analysis, a MCPR operating limit of 1.15 for WNP-2 with a MCPR safety limit of 1.06 is supported (i.e., a change in CPR of 0.09). The WNP-2 MCPR safety limit for Cycle 5 continues to be 1.06; hence the LOFH transient requires a MCPR operating limit of 1.15 for WNP-2 (Reference 2.0).

9.2 Local Transients

Analyses given in Reference 1.0 show that the FLE transient is bounded by the CRWE transient and is therefore nonlimiting. Based on the CRWE results, the MCPR operating limit is a function of the RBM setpoint. Analyses were performed to support a RBM setpoint of 106 percent, 107 percent, and 108 percent. The change in CPR for the CRWE with a 106 percent RBM setpoint is 0.17 for ANF fuel and 0.17 for GE fuel, for a 107 percent RBM setpoint 0.19 for ANF fuel and 0.19 for GE fuel, and for a 108 percent RBM setpoint 0.21 for ANF fuel, and 0.21 for GE fuel.

9.3 Reduced Flow Operation

The recirculation flow run-up analysis performed for WNP-2 Cycle 2 was reviewed and was found to change due to a change of model input data influenced by operational data. The reduced flow MCPR operating limit for WNP-2 Cycle 5 is found to be more conservative than the previous reduced flow MCPR operating limit. For final feedwater temperature reduction (FFTR) conditions, the previously reported reduced flow MCPR operating limit remains applicable. The applicable limit for Cycle 5 is given in Table 9.1 (Reference 2.0).

Table 9.1

Reduced Flow MCPR Operating Limit For Cycle 5, WNP-2

Core Flow (% rated)	Reduced Flow MCPR Operating Limit
100	1.07
90	1.13
80	1.19
70	1.26
60	1.34
50	1.44
40	1.59

9.4 ASME Overpressurization Analysis

In order to demonstrate compliance with the ASME Code over pressurization criteria of 110 percent of vessel design pressure (1375 psig), the Main Steam Isolation Valve (MSIV) closure event with failure of the MSIV position switch scram was analyzed with ANF's COTRANSA code (Reference 11.0). The WNP-2 Cycle 5 analysis assumed six safety relief valves out of service. The maximum pressure observed in the analysis is 1315 psig in the vessel lower plenum. This is 105 percent of the reactor vessel design pressure which is below the 110 percent design criterion of 1375 psig.

The calculated steam dome pressure corresponding to the 1315 psig peak vessel pressure is 1286 psig, for a vessel differential pressure of 29 psig. The RPT is assumed to initiate at a pressure setpoint of 1170 psig. The current Technical Specification Safety limit of 1325 psig is based on dome pressure and therefore conservatively assumes a 50 psi vessel pressure differential (1375-1325). Since the calculated vessel differential pressure is 29 psi, the steam dome safety limit of 1325 psig assures compliance with the ASME criterion of 1375 psig peak vessel pressure.

9.5 Increased Flow Operation

The plant system transient events reported earlier in this document, which are potentially limiting for MCPR, were all analyzed at increased core flow of 106 percent. The Cycle 2 transient events analyzed at the design basis power condition with increased core flow were found to bound the same transients analyzed at the design basis power and rated flow condition for WNP-2 Cycle 2 (Reference 14.0).

ANF has also performed analyses which demonstrate that the XN-1 8x8C fuel bundle can operate satisfactorily from a mechanical standpoint at this increased core flow (Reference 15.0). In addition, GE has performed analyses for the reactor internals and for the GE fuel assembly which considered the loads created by operation at this flow level and the impacts of these loads on the WNP-2 core internals and the GE fuel assembly. Also, flow induced vibration of the core internals as a result of increased core flow was analyzed. Finally, analyses were performed for feedwater nozzle and feedwater sparger fatigue at increased core flow (Reference 16.0). The results of all these analyses when considered along with the similarity with the fuel types utilized in Cycle 5, confirm the capability of WNP-2 to operate at 100 percent power and 106 percent core flow during Cycle 5 operation. A review of the 9x9 LFA fuel, discussed in Section 6.0, confirms its capability for increased flow operation.

A containment analysis was performed to determine the impact of operation at increased core flow on the WNP-2 containment LOCA response. The results show that the containment LOCA response for increased core flow operation is bounded by the corresponding FSAR results (Reference 17.0).

In summary, all relevant neutronic, thermal hydraulic, mechanical, and safety analyses have been performed to demonstrate that WNP-2 can operate safely with extended core flow up to 106 percent of rated core flow during Cycle 5.

9.6 Single Loop Operation

ANF performed analyses for WNP-2 which demonstrate the safety of plant operation with a single recirculation loop out of service at 75% of rated power for an extended period of time. These analyses were performed for the most limiting transient events, the pump seizure accident and the loss-of-coolant-accident (LOCA) for the maximum extended power state during WNP-2 single loop operation (SLO). The results of the SLO analyses are summarized below:

- o The two loop MCPR operating limits (rated conditions) bound the transient requirements for SLO. The single loop transient analyses need not be performed on a cycle by cycle basis and a MCPR = 1.35 is appropriate for single loop conditions.

- o The postulated pump seizure accident, evaluated for SLO conditions, is calculated to have a less severe radiological release than the LOCA. The radiological consequences of this postulated accident are bounded by the radiological evaluation performed by GE for the LOCA and are well within the 10CFR100 limits.

- o The single loop ECCS analysis supports the use of the WNP-2 two loop MAPLHGR limits for ANF fuel when the reactor is operating in the SLO mode consistent with the single loop MCPR Operation limit (1.35 at 50 percent of rated flow). Single loop operation of WNP-2 with the two loop ANF fuel MAPLHGR limits assures that the emergency core cooling systems for the WNP-2 plant will meet the U.S. NRC acceptance criteria of 10CFR50.46 for loss-of-coolant accident breaks up to and including the double-ended severence of a reactor coolant pipe.

The transient and pump seizure accident analyses are described as Reference 18.0 and the LOCA analyses are described in Reference 19.0.

With a single recirculation loop in operation, the GE analyses supported continued operation with an increase of 0.01 in the MCPR safety limit. ANF performed a single loop MCPR safety limit calculation and found that less than one tenth of one percent of the rods to be in boiling transition which supports a MCPR safety limit of 1.07. Because of the similarity between the ANF and GE fuel types making up the core, and because of the similarity in the magnitude of the uncertainties which determine the MCPR safety limit, this small increase in the safety limit value can be used for operation with ANF fuel and single loop analyses. For Cycle 5 operation with both recirculation loops in operation, the MCPR safety limit is 1.06, which is the same value as was used for the previous cycles. For Cycle 5 operation with a single recirculation loop in service, the MCPR safety limit is 1.07, which is also the same value used for the previous cycles. The LFA single loop operation limits are bounded by the two-loop operation limits (Reference 1.0).

9.7 Final Feedwater Temperature Reduction

Reference 20.0 presents a final feedwater temperature reduction (FFTR) analysis with thermal coastdown for WNP-2. This analysis is the subject of a currently proposed Tech. Spec. change and is applicable to future WNP-2 fuel cycles. The FFTR analysis was performed for a 65°F temperature reduction. This FFTR analysis is applicable after the all rods out condition is reached with normal feedwater temperature. The FFTR analysis results show that CPR changes for the LRNB and FWCF transients of + 0.02 and - 0.01, respectively, are applicable to these respective anticipated operational occurrence (AOO) events. That is, these LRNB and FWCF limit changes are applicable when Cycle 5 reactor operation is being extended with thermal coastdown at FFTR conditions and are applicable to both the 8x8 and 9x9 reload fuel designs (Reference 2.0).

10.0 POSTULATED ACCIDENTS

For Cycle 2, ANF had analyzed the LOCA to determine MAPLHGR limits for ANF 8x8 fuel. The results of this analysis are presented in Reference 21.0. For the 9x9 LFA fuels, these same MAPLHGR limits as modified to account for the difference in activated fuel pin length between the two fuel types, are also applicable to the 9x9 LFA fuel (Reference 1.0). These Cycle 2 results are equally applicable to Cycle 5. ANF's methodology for the LOCA analysis is given in References 22.0, 23.0, and 24.0. In addition, the Rod Drop Accident (RDA) was analyzed to demonstrate compliance with the 280 cal/gm design limit. ANF's methodology for the RDA analysis can be found in Reference 10.0.

10.1 Loss of Coolant Accident

Reference 25.0 describes ANF's WNP-2 LOCA break spectrum analysis which defined the limiting break for WNP-2. The analysis of this event for WNP-2 is described in Reference 26.0. The LOCA analysis described in Reference 26.0 was performed for an entire core of ANF 8x8C fuel and therefore provides MAPLHGR limits for ANF fuel only. These results are applicable to operation in WNP-2 Cycle 5 including the 9x9 LFA fuel (Reference 1.0).

ANF reload fuel is hydraulically and neutronically compatible with the GE initial core fuel. Therefore, the existing GE LOCA analysis and MAPLHGR limits are applicable to GE initial core fuel during Cycle 5 operation.

10.2 Rod Drop Accident

ANF's methodology for analyzing the RDA is given in Reference 10.0. For WNP-2 Cycle 5, the analysis shows a value of 121 cal/gm for the maximum deposited fuel rod enthalpy during the worst case postulated RDA (Reference 1.0). This is well below the design limit value of 280 cal/gm.

10.3 Single Loop Operation

To support operation of WNP-2 with a core composed of GE Cycle 1 fuel and ANF reload fuel with a single recirculation pump operating, ANF recommends the conservative use of GE MAPLHGR limits for the GE fuel design with a multiplier of 0.84 applied for single loop operation. The single loop ECCS analysis supports the use of the WNP-2 two loop MAPLHGR limits for ANF fuel when the reactor is operating in the SLO mode. Single loop operation of WNP-2 with the two loop ANF fuel MAPHGR limits assures that the emergency core cooling systems for the WNP-2 plant will meet the U.S. NRC acceptance criteria of 10CFR50.46 for loss-of-coolant accident breaks up to and including the double-ended severance of a reactor coolant pipe.

11.0 STARTUP PHYSICS TEST PROGRAM

The Supply System has developed a restart physics test program to be carried out prior to or during Cycle 5 startup. This program includes a core loading verification test, a control rod functional test, an in sequence shutdown margin test, and a TIP asymmetry test. The proposed test goals and a brief description of each test is given below.

11.1 Core Load Verification Test

Goal - To assure that the WNP-2 Cycle 5 Core is loaded according to the design analyzed by ANF.

Test Description - This test will be performed with the aid of a television camera. A series of initial passes will be made with the fuel mast set at a predetermined height to assure that all fuel assemblies are fully seated in the core. Then, with the aid of the camera and a visual readout on the refuel floor, the assembly serial numbers, their orientation and location will be visually checked and recorded on video tape. Subsequently, a review of the tapes will be made to check the initial verification.

11.2 Control Rod Functional Test

Goal - To determine and verify control rod mobility and functionality.

Test Description - Following the completion of fuel loading, for each cell of four fuel assemblies, the control·blade for that cell will be fully withdrawn and inserted. This will demonstrate the mobility of that blade, the absence of overtravel for that blade and the fact that the lattice is subcritical with that blade withdrawn. This in turn will verify that there are no gross reactivity discrepancies between the actual core and the analyzed design.

After the core is fully loaded, verify that the control rod drive insertion and withdrawal times are within design specifications and technical specification limits. This action will also verify that the core is subcritical with any single rod fully withdrawn.

11.3 Subcritical Margin Test

Goal - To assure that the Technical Specification shutdown margin requirement is satisfied.

Test Description - The data is taken during a normal insequence startup criticality. Critical control rod positions are obtained and corrected for reactor period and moderator temperature coefficient effects. The results are compared to predicted control rod positions and from this information, the shutdown margin with the analytically determined strongest control rod withdrawn is confirmed.

11.4 TIP Asymmetry Test

Goal - To assure proper TIP systems operation and to verify that the TIP system uncertainty is within the limits assumed for transient analysis.

Test Description - This test is performed in the power range preferably above 75 percent power. An octant symmetric control rod pattern is utilized. Data is gathered from all available TIP locations, and the total average uncertainty is determined for all symmetric TIP pairs.

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