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WNP-2 CYCLE 3 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 partial support of the WNP-2 Cycle 3 reloading licensing submittal. The information contained herein is true and correct to the best of the Supply System's knowledge, information, and belief.

WNP-2 CYCLE 3 RELOAD SUMMARY REPORT

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WNP-2 CYCLE 3 RELOAD SUMMARY REPORT

1.0 INTRODUCTION

The second reload of the Washington Public Power Supply System Plant No. 2 (WNP-2) will utilize Advanced Nuclear Fuels Corporation (ANF), (formerly Exxon Nuclear Company (ENC)), 8x8 current fuel. The fuel design of this reload batch is virtually identical to the fuel design of the previous reload batch. This report summarizes the reload analyses performed by ANF in support of WNP-2 operation for Cycle 3. In addition, a description of the ANF reload is given along with a comparison of the characteristics of the Cycle 3 and Cycle 2 cores. A discussion of the proposed physics startup program is also included. The proposed license amendment (technical specification changes) are listed by title in this report for completeness.

The reload licensing submittal is composed on the WNP-2 Cycle 3 Reload Analysis Report (XN-NF-87-25) (Reference 1.0), the WNP-2 Cycle 3 Plant Transient Analysis Report (XN-NF-87-24) (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 3 Reload Analysis Report (Reference 1.0) is intended to be used in conjunction with ENC Topical Report XN-NF-80-19(P)(A), Volume 4, Revision 1, Application of the ENC 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

For the second refueling outage for WNP-2, the Supply System will replace 148 of the General Electric (GE) initial core fuel assemblies with ANF 8x8C fuel. The 148 ANF 8x8C fuel bundles to be loaded for Cycle 3 (Reference 4.0) are similar in design to the initial core fuel. However, the change in WNP-2 core loading requires a re-analysis by ANF. Much of this analysis, particularly the Loss of Coolant Accident (LOCA) and the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) are given in Reference 4.0 as these analyses were performed for all ANF fueled cores as a part of the Cycle 2 analysis. Relevant transient analyses and Minimum Critical Power Ratio (MCPR) analyses for the Cycle 3 loading are reported here. Analyses of normal reactor operation consisted of evaluation of the mechanical, thermal hydraulic, and nuclear design characteristics. Operation at extended core flow is 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 3 core. A list of these Technical Specification changes is given in Table 2.1.

TABLE 2.1

PROPOSED TECHNICAL SPECIFICATION CHANGES

2.0	Safety Limits and Limiting Safety System Setting (Introduction)
2.1.2	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
3/4.3.10	Neutron Flux Monitoring Instrumentation
B 3/4.1.3	Control Rods
B 3/4.2.1	Average Planar Linear Heat Generation Rate
B 3/4.2.3	Minimum Critical Power Ratio
B 3/4.7.9	Main Turbine Bypass Systems

3.0 WNP-2 CYCLE 3 OPERATING HISTORY

WNP-2, a 3323 mwt BWR 5, began Cycle 2 operation on June 10, 1986. The end of Cycle 2 operation is expected to be April 13, 1987.

During Cycle 2, the plant was base loaded at or near 100 percent power for the first five months of the cycle. At this point, excessive vibration was observed in Recirculation Pump A. This pump was shut down at that point and operation continued in single loop. In single loop operation, WNP-2 is limited to approximately 72 percent power. WNP-2 will continue to operate in single loop for the remainder of Cycle 2.

Figure 3.1 gives a power history of Cycle 2 through March 12, 1987, for WNP-2. The Cycle 2 operating highlights and control rod sequence exchange schedule are found in Table 3.1.

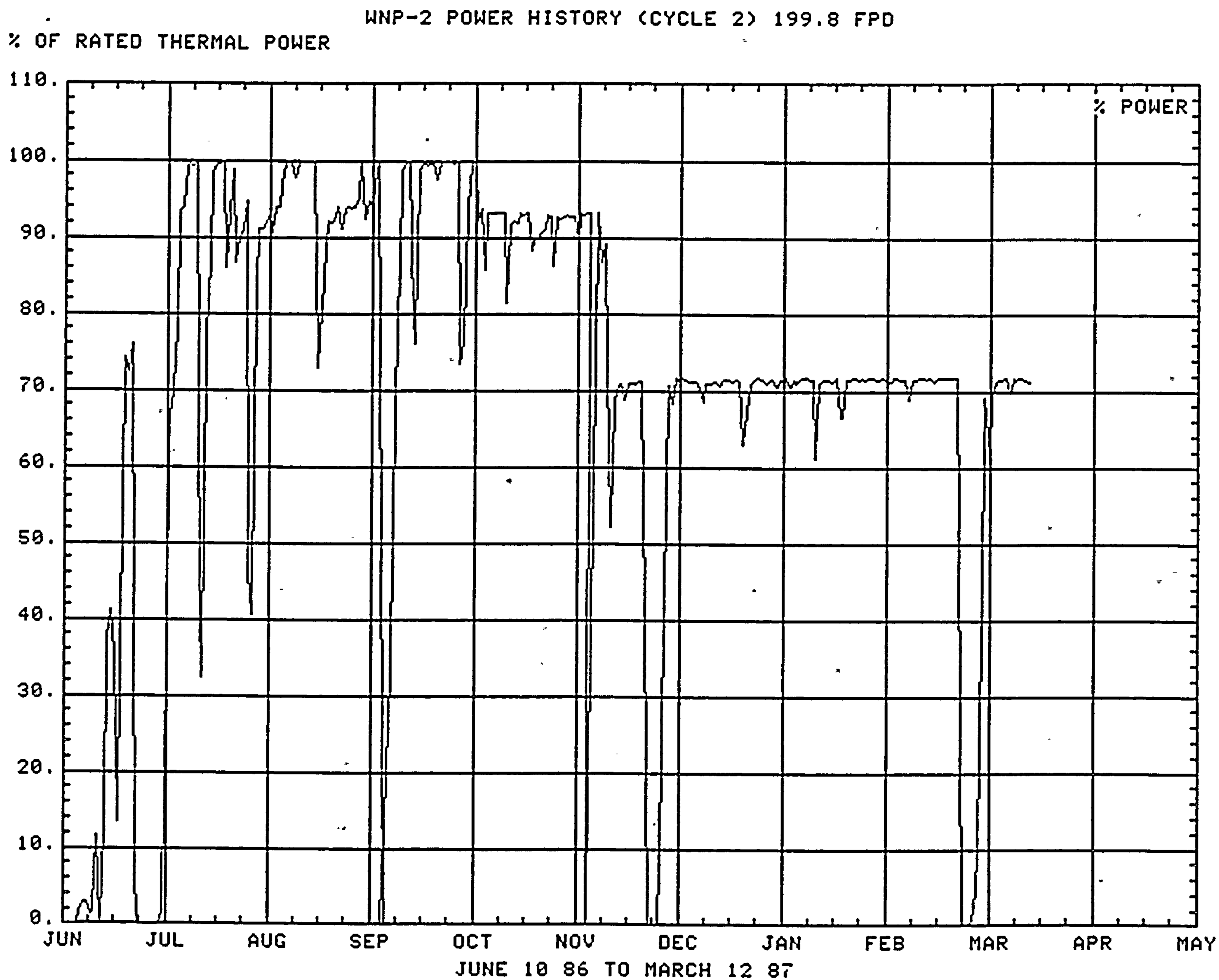

Figure 3.1 Power History For WNP-2 For Cycle 2

TABLE 3.1

WNP-2 CYCLE OPERATING HIGHLIGHTS

Began Fuel Loading	April 18, 1986
Began Commercial Operation	June 10, 1986
Projected End of Cycle Date	April 13, 1987
End of Cycle Core Average Exposure (Design)(mwd/mtm)	12,153
Number of Fresh Assemblies	128
Gross Generation (FPD) (through March 18, 1987)	204.1

Control Rod Sequence Exchange Schedule

Date	Sequence From	Sequence To
August 14, 1986	A2	B2
September 26, 1986	B2	A1
November 10, 1986	A1	B1
January 10, 1987	B1	A2
February 26, 1987	A2	A1

4.0 RELOAD CORE DESCRIPTION

The WNP-2 core consists of 764 fuel assemblies. For the Cycle 3 reload, the core will consist of 148 ANF 8x8C fresh assemblies, 128 ENC 8x8C XN-1 fuel assemblies loaded for Cycle 2 and 488 GE 8x8RP assemblies remaining from the initial core. The 148 ANF 8x8C fresh assemblies consist of 36 reload assemblies originally manufactured for loading in Cycle 2 and 112 reload assemblies manufactured for loading in Cycle 3. The two assemblies are identical in uranium dioxide (UO_2) enrichment, gadolinium oxide (GD_2O_3) loading, and in all other major physical characteristics. Minor differences, primarily in end plug design, exist between the two assembly designs. However, the two 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 3 core.

TABLE 4.1

WNP-2 CYCLE 3 CORE

Number of Assemblies	Type	Enrichment
148*	ANF 8x8C	2.72 w/o U-235
128**	ENC 8x8C	2.72 w/o U-235
432	GE 8x8RP	2.19 w/o U-235
56	GE 8x8RP	1.76 w/o U-235

The 148 exposed GE 8x8RP assemblies discharged are all medium enriched (1.76 w/o U-235) assemblies.

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

**Two of these assemblies are Lead Test Assemblies (LTA) described in Reference 4.0.

5.0 FUEL MECHANICAL DESIGN

The mechanical design of the 8x8C Cycle 3 ANF reload fuel for WNP-2 is described specifically in Reference 5.0 and more generically in Reference 6.0 and 7.0. 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 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 Cycle 3 reload fuel evaluated the following items in 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 work 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 Cycle 3 ANF reload fuel is the same as the structural response of the 8x8C Cycle 2 ENC fuel and the 8x8RP GE fuel which also reside in the WNP-2 core. As a part of Cycle 3 operation, some of the 8x8C Cycle 3 ANF reload fuel assemblies will be channeled with new 100 mil channels fabricated by ASEA Atom. These channels are equivalent to the initial core channels. 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 occurence of the design basis seismic LOCA event.

A LHGR limit will be placed on ANF 8x8C Cycle 3 reload fuel assemblies for monitoring for the reasons given previously in Reference 4.0, Page 10, for ENC 8x8C Cycle 2 fuel.

6.0 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.

6.1 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 two fuel designs and their measured performance characteristics demonstrate that the two fuel designs are sufficiently compatible for co-residence in WNP-2.

6.2 Fuel Cladding Integrity Safety Limit

The MCPR fuel cladding integrity safety limit for WNP-2 is 1.06 which is equal to the Cycle 1 and Cycle 2 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 3 MCPR safety limit analysis methodology and input parameters are described in XN-NF-87-24, Cycle 3 Plant Transient Report (Reference 2.0).

6.3 Fuel Centerline Temperature

The LHGR curve in Figure 3.4 of Reference 8.0 shows that the ANF 8x8C fuel centerline temperature is protected for 120 percent over power. The LHGR curve in Reference 8.0 is everywhere greater than 120 percent of the LHGR limit curve in Reference 6.0. Therefore, fuel centerline melt is protected for all ANF 8x8 exposures within the bounds of the referenced LHGR curve.

6.4 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 3 is 11.6 percent of the total core flow which is similar to the Cycle 1 value of 11.8 percent and identical to the Cycle 2 value of 11.6 percent. The computed bypass flow will have no adverse impact on reactor operation.

6.5 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 (48 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 11.0).

7.0 NUCLEAR DESIGN

The neutronic methods for the design and analysis of the WNP-2 Cycle 3 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.

7.1 Fuel Bundle Nuclear Design

The Cycle 3 ANF reload bundles are identical to the Cycle 2 ANF reload bundles in nuclear design. Major nuclear design characteristics for the ANF 8x8C reload fuel assembly are:

- The fuel assembly contains 62 fuel rods and two water rods. One of the water rods also acts as a spacer capture rod.
- The fuel assembly average enrichment is 2.72 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 2.89 w/o U-235.
- Five enrichment levels are utilized in the 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.
- Each fuel assembly contains five fuel rods with 2.0 w/o GD_2O_3 blended with 2.57 w/o U-235 enriched UO_2 to reduce initial assembly reactivity.

The enrichment distribution of the ANF reload design 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 assembly, three rods are enriched to 1.5 w/o U-235, seven rods to 2.0 w/o U-235, nine rods to 2.57 w/o U-235, 16 rods to 2.94 w/o U-235, 22 rods to 3.54 w/o U-235, and five rods to 2.57 w/o U-235 plus 2.00 w/o GD_2O_3.

7.2 Core Nuclear Design

The core exposure for the end of Cycle 2 (EOC2), the core exposure for the beginning of Cycle 3 (BOC3), and the core exposure for the end of Cycle 3 (EOC3) 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. Some of the results of these analyses are shown in Table 7.1.

Table 7.1

CORE NUCLEAR DESIGN

Core Exposures at EOC2 (mwd/mtm)	12,153
Core Exposures at BOC3 (mwd/mtm)	9,639
Core Exposures at EOC3 (mwd/mtm)	15,103
BOC Cold K_{eff}, all rods out	1.1257
BOC Cold K_{eff}, strongest rod out	0.9882
Reactivity Defect/R-Value, percent Δ K/K	0.0
Standby Liquid Control System (SBLC) Reactivity, 660 PPM Boron, K_{eff}	0.9722

7.3 Comparison of Major Core Parameters

Some of the major core parameters for WNP-2 Cycle 2 and Cycle 3 are listed in Table 7.2.

Table 7.1

COMPARISON OF MAJOR CORE PARAMETERS

Parameter	Cycle 2	Cycle 3
MCPR Limit (0 mwd/mtm)	1.28	1.29
Doppler Defect ($\Delta K_\infty / K_\infty \ \Delta T$)	$- 9.5 \times 10^{-6}$	$- 9.5 \times 10^{-6}$
Cycle Length (Design; FPD)	255	227
Core Average Exposure (BOC; mwd/mtm)	7,424	9,639
Core Average Exposure (EOC; mwd/mtm)	12,153	15,103

The differences between the Cycle 2 core and the Cycle 3 core are found in the core loading pattern. The Cycle 2 core consisted of a scatter load of 204 GE 8x8 medium (1.76 w/o U-235) and 432 high enriched (2.19 w/o U-235) bundles and 128 ANF 8x8 reload (2.72 w/o U-235) bundles. The Cycle 3 core will consist of a scatter load of 52 GE 8x8R medium enriched bundles, 436 GE 8x8R high enriched bundles, 128 ANF 8x8C reload bundles with one cycle of exposure and 148 ANF 8x8C fresh reload bundles.

8.0 ANTICIPATED OPERATIONAL OCCURRENCES

ANF considers eight categories of potential system core wide transient occurrences for jet pump BWRs in Reference 12.0. ANF has provided analysis results for the three most limiting transients for WNP-2 Cycle 3 to determine the Cycle 3 thermal margins. The three transients determined to be most limiting for Cycle 3 are:

- o Load Rejection Without Bypass (LRWB).
- o Feedwater Controller Failure (FWCF).
- o Loss of Feedwater Heating (LOFH).

ANF's methodology for developing thermal limits is found in Reference 13.0. Reference 12.0 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 results of the core-wide and local transient analyses are provided in the WNP-2 Cycle 3 Reload Analysis Report (Reference 1.0) and in the WNP-2 Cycle 3 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 Δ CPR of 0.20 for the ANF fuel and 0.23 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 1.26 for the ANF fuel and 1.29 for the GE fuel in Cycle 3 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 FWCF transient was determined to be the limiting transient and the MCPR limit for ANF fuel is 1.30 and for GE fuel is 1.32 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 3/4.1.7) for this portion of the fuel cycle (EOC-2000 mwd/mtm to EOC). For operation at EOC-2000 with core flow up to 106 percent and these technical specification scram times, the limiting transient is the

LRWB transient and the MCPR operating limit within EOC-2000 mwd/mtm to EOC is 1.35 for ANF fuel and 1.39 for GE fuel for Cycle 3 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, the limiting transient is then the FWCF transient and the MCPR operating limit is 1.35 for ANF fuel and 1.37 for GE fuel. Finally, if the RPT becomes inoperable within EOC-2000 mwd/mtm to EOC and the plant defaults to technical specification scram times, the LRWB transient at 106 percent flow is bounding and the MCPR operating limit is 1.39 for ANF fuel and 1.43 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 Δ CPRs are 0.20 for ANF fuel and, 0.23 for GE fuel at 107 percent, 0.22 for ANF fuel, and 0.25 for GE fuel at a 108 percent rod block setting. Therefore, operation with a 108 percent RBM setting would require a MCPR limit of 1.28 for ANF and 1.31 for the GE fuel.

8.1 Core Wide Transients

The plant transient model used to evaluate the pressurization transients, the LRWB and FWCF events, consists of the ANF COTRANSA (Reference 12.0) and XCOBRA-T (Reference 14.0) 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 FWCF 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 LRWB 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 FWCF 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 LRWB 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) transient was analyzed on a generic basis for a wide cross section of BWR configurations. This generic analysis is documented in Reference 15.0. This analysis provides a statistical evaluation of the consequences of the LOFH transient for BWR/4, BWR/5, and BWR/6 plant configurations under conditions which cover the operating power flow map including increased core flow conditions. At this time, the generic analysis is under review. A conservative bounding value of a Δ CPR of 0.09 is supported by the analysis results for plants with a MCPR safety limit of 1.06. The WNP-2 MCPR safety limit for Cycle 3 continues to be 1.06 (Reference 2.0). Therefore, the LOFH transient requires a MCPR operating limit of 1.15 for WNP-2.

8.2 Local Transients

Analysis 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 Δ CPR for the CRWE with a 106 percent RBM setpoint is 0.20 for ANF fuel and 0.23 for GE fuel, for a 107 percent RBM setpoint 0.20 for ANF fuel and 0.23 for GE fuel, and for a 108 percent RBM setpoint 0.22 for ANF fuel, and 0.25 for GE fuel.

8.3 Reduced Flow Operation

The recirculation flow run-up analysis performed for WNP-2 Cycle 2 was reviewed and the assumptions and conditions used for Cycle 2 are applicable to Cycle 3. Thus, the reduced flow MCPR operating limit for WNP-2 Cycle 2 is applicable to Cycle 3.

8.4 ASME Overpressurization Analysis

In order to demonstrate compliance with the ASME Code over pressurization criteria of 110 percent of vessel design pressure, the Main Steam Isolation Valve (MSIV) closure event with failure of the MSIV position switch scram was analyzed with ANF's COTRANSA code (Reference 12.0). The WNP-2 Cycle 3 analysis assumed six safety relief valves out of service. The maximum pressure observed in the analysis is 1313 psig in the vessel lower plenum. This is 105 percent of the reactor vessel design pressure which is well below the 110 percent design criterion.

The calculated steam dome pressure corresponding to the 1313 psig peak vessel pressure is 1285 psig, for a vessel differential pressure of 28 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 dp (1375-1325). Since the calculated vessel differential pressure is 28 psi, the steam dome safety limit of 1325 psig assures compliance with the ASME criterion of 1375 psig peak vessel pressure.

8.5 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 16.0).

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 17.0). 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. The results of all these analyses when considered along with the similarity between the two fuel types utilized in Cycle 3, confirm the capability of WNP-2 to operate at 100 percent power and 106 percent core flow during Cycle 3 operation (Reference 18.0).

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 19.0).

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 3.

8.6 Single Loop Operation

The NSSS Supplier, GE, has provided analyses which demonstrate the safety of WNP-2 operation with a single recirculation loop in service for an extended period of time (Reference 20.0). Because the ANF fuel is designed to be compatible with the GE fuel and because the ANF methodology gives results consistent with GE for two loop operation, the GE single loop analysis is also applicable to ANF 8x8 fuel. With a single loop in operation, the GE analysis supported operation with an increase of 0.01 in the MCPR safety limit. Due to compatibility, this increase is also appropriate for ANF 8x8 fuel. For Cycle 3 operation of WNP-2 with a single loop in service, the MCPR safety limit is 1.07 which is the same value as the previous cycle.

The consequences of core wide transients for single loop operation are bounded by the consequences of these events at rated conditions.

The additional conservatism imposed by the reduced flow MCPR operating limits assure that the MCPR safety limit will not be violated during single loop operation. Because the reduced flow MCPR limit curves are based on equipment performance which cannot physically happen during single loop operation, the added conservatism present in the curves compensates for the penalties associated with increased uncertainties in the MCPR limit and control rod drive performance. The reduced flow MCPR limit curves are applicable without modification during single loop operation.

9.0 POSTULATED ACCIDENTS

For Cycle 2, ANF has analyzed the LOCA to determine MAPLHGR limits for ANF 8x8 fuel. The results of this analysis is presented Reference 21.0. These results are equally applicable to Cycle 3. 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.

9.1 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 3.

ANF 8x8C 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 3 and future cycles with mixed GE/ANF cores.

9.2 Rod Drop Accident

ANF's methodology for analyzing the RDA is given in Reference 10.0. For WNP-2 Cycle 3, the analysis shows a value of 170 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.

9.3 Single Loop Operation

To support operation of WNP-2 with a core composed of GE Cycle 1 fuel and ANF 8x8 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 analytical limits used by GE have yielded conservative MAPLHGR limits relative to the MAPLHGR limits obtained using the approved ANF analytical methods. The phenomena which requires the reduction in MAPLHGR limits for single loop operation are common to both fuels. Therefore, applying the more conservative GE MAPLHGR limits to ANF fuel assures conformance with the criteria of 10CFR50.46.

10.0 STARTUP PHYSICS TEST PROGRAM

The Supply System has developed a restart physics test program to be carried out prior to initiation of Cycle 3. This program includes a core loading verification test, a control rod functional test, a in sequence shutdown margin test, and a TIP asymmetry test. The proposed test goals and a brief description of each test is given below.

10.1 Core Load Verification Test

Goal - To assure that the WNP-2 Cycle 3 Core is loaded according to the design analyzed by ANF.

Test Description - This test will be performed with the aid of a television camera mounted on the fuel mast. A series of initial passes will be made with the television camera/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.

10.2 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.

10.3 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.

10.4 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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