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SUBJECT: Application for amend to License NPF-21, consisiting of changes to Tech Spec 3/4.2.3, "Min Critical Power Ratio" & 3/4.3.10, "Neutron Flux Monitoring Instrumentation." Fee paid.

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Washington Public Power Supply System

3000 George Washington Way P.O. Box 968 Richland, Washington 99352-0968 (509)372-5000

March 7, 1988
G02-88-054
Docket No. 50-397

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Subject: NUCLEAR PLANT NO. 2
OPERATING LICENSE NPF-21
REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS -
RELOAD LICENSE AMENDMENT (CYCLE 4)

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, the Supply System hereby requests an amendment to the WNP-2 Technical Specifications (Tech. Specs.). This amendment is being submitted to allow the use of Advanced Nuclear Fuels Corporation Inc. (ANF) reload fuel in Cycle 4 of WNP-2. Changes to the following Tech. Specs. are being requested.

3/4.2.3 Minimum Critical Power Ratio

3/4.3.10 Neutron Flux Monitoring Instrumentation

The following attachments to this letter are a Reload Summary Report, marked up Tech. Specs. and the plant specific documents generated by ANF, all of which provide the basis for the proposed no significant hazards determination.

I. Technical Specification Changes

II. WNP-2 Cycle 4 Reload Summary Report (WPPSS-EANF-119). (Includes the Startup Physics Program).

III. WNP-2 Cycle 4 Reload Analysis (XN-NF-88-02).

IV. WNP-2 Cycle 4 Plant Transient Analysis (XN-NF-88-01).

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REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS
RELOAD LICENSE AMENDMENT (CYCLE 4)

The Supply System has reviewed the use of ANF reload fuel in Cycle 4 of WNP-2 and concludes that it does not involve an unreviewed safety question. The Supply System has also evaluated this request per 10CFR50.92 and provides the following in support of the finding for no significant hazards considerations.

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because the transient analyses have been reanalyzed for the reload core. The proposed change to the Tech. Specs. reflects new operating limits associated with the reload core, are based on approved analysis methods and are within the current acceptance criteria.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because the operational limitations applied to Cycle 4 are identical to previous cycles. The values were derived from NRC qualified codes and by applying the most limiting transients throughout the cycle. These limitations are sufficient to ensure the plant is operated within previously accepted conditions. In addition, no changes sufficient to create a new type of malfunction are contemplated.

3) Create a significant reduction in the margin of safety because the margin to safety for all accidents or operational occurrences analyzed for Cycle 4 operation is either identical to or more conservative than that used for previous cycles.

As discussed above, the Supply System considers that this change does not involve a significant hazards consideration, nor is there a potential for significant change in the types or significant increase in the amount of any effluents that may be released offsite, nor does it involve a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9) and therefore, per 10CFR51.22(b), an environmental assessment of the change is not required.

REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS
RELOAD LICENSE AMENDMENT (CYCLE 4)

This Technical Specification change has been reviewed and approved by the WNP-2 Plant Operations Committee (POC) and the Supply System Corporate Nuclear Safety Review Board (CNSRB).

In accordance with 10CFR170.21, an application fee of one hundred fifty dollars ($150.00) accompanies this request. In accordance with 10CFR50.91, the State of Washington has been provided a copy of this letter.

WNP-2 is scheduled to begin the spring outage on April 25, 1988. The plant is currently scheduled to resume commercial operation on or about June 1, 1988.

Should you have any questions, please contact the Manager, WNP-2 Licensing.

Very truly yours,

R B Glasscock
for G. C. Sorensen, Manager
Regulatory Programs

HLA/WCW/lw

Attachments

cc: C Eschels - EFSEC
JB Martin - NRC RV
NS Reynolds - BCP&R
RB Samworth - NRC
DL Williams - BPA/399
NRC Site Inspector - 901A

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Amend to TS - Reload License

I, R. B. GLASSCOCK, being duly sworn, subscribe to and say that I am the Director, Licensing and Assurance, for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that I have full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information and belief the statements made in it are true.

DATE March 4, 1988

RB Glasscock

R. B. GLASSCOCK, Director
Licensing and Assurance

On this day personally appeared before me R. B. GLASSCOCK to me known to be the individual who executed the foregoing instrument and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 4th day of March, 1988.

Bernice Kosko

Notary Public in and for the
State of Washington

Residing at Kennewick

8803150282

WNP-2 CYCLE 4 RELOAD SUMMARY REPORT

Prepared By: ______________________________
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Concur With: KD Cowan 2/24/88
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Approved By: ______________________________ 2/25/88
D. L. Larkin, Manager, Engineering Analysis and Nuclear Fuel

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 Application For Technical Specifications Changes Relating to WNP-2 Cycle 4 operation. The information contained herein is true and correct to the best of the Supply System's knowledge, information, and belief.

WNP-2 CYCLE 4 RELOAD SUMMARY REPORT

TABLE OF CONTENTS

TABLE OF CONTENTS (CONTD.)

WNP-2 CYCLE 4 RELOAD SUMMARY REPORT

1.0 INTRODUCTION

The third reload of the Washington Public Power Supply System Plant No. 2 (WNP-2) will utilize Advanced Nuclear Fuels Corporation (ANF), 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 4. In addition, a description of the ANF reload is given along with a comparison of the characteristics of the Cycle 4 and Cycle 3 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 of the WNP-2 Cycle 4 Reload Analysis Report (XN-NF-88-02) (Reference 1.0), the WNP-2 Cycle 4 Plant Transient Analysis Report (XN-NF-88-01) (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 4 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

For the third refueling outage for WNP-2, the Supply System will replace 152 of the General Electric (GE) initial core fuel assemblies with ANF 8x8C fuel. Twenty-four (24) of the Cycle 4 reload fuel assemblies will have a bundle average enrichment of 2.72 weight percent U^{235}; 128 of the Cycle 4 reload fuel assemblies are enriched to a bundle average value of 2.64 weight percent U^{235}. The 152 ANF 8x8C fuel bundles to be loaded for Cycle 4 (Reference 4.0) are similar in design to the initial core fuel and previous reload assemblies. However, the change in WNP-2 core loading requires a partial re-analysis by ANF. The Loss of Coolant Accident (LOCA) and the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) relevant to Cycle 4 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. Relevant transient analyses and Minimum Critical Power Ratio (MCPR) analyses for the Cycle 4 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, 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 4 core. A list of these Technical Specification changes is given in Table 2.1.

TABLE 2.1

PROPOSED TECHNICAL SPECIFICATION CHANGES

3/4.2.3	Minimum Critical Power Ratio
3/4.3.10	Neutron Flux Monitoring Instrumentation

2.1 SUMMARY OF RESULTS

The limiting transient MCPR results for the analyses described in sections 7 and 8 of this document are summarized in columns 3 and 4 of Table 2.2.

WNP-2 will be entering its fourth cycle of operation and is approaching an equilibrium cycle. Analysis results between cycle 3 and 4 show little change. As a result, WNP-2 has chosen to add some small CPR penalties for margin to envelope future anticipated analysis results. The intent is to be able to submit future reload applications which require no Technical Specification changes, thereby allowing application per the provisions of 10CFR50.59. The resulting CPR limits, including these self-imposed penalties, are summarized in columns 5 and 6 of Table 2.2.

TABLE 2.2

MCPR OPERATING LIMITS

1. Cycle Explosure	2. Equipment Status	3. Analysis GE Fuel	4. Results ANF Fuel	5. Submittal GE Fuel	6. Values ANF Fuel
0-3750 MWD/MT	NA	1.29	1.26	1.40	1.28
3750 - EOC	Normal SCRAM times	1.31	1.30	1.40	1.31
3750 - EOC	TS SCRAM times	1.38	1.36	1.50	1.38
3750 - EOC	RPT inop Normal SCRAM times	1.38	1.35	1.50	1.37
3750 - EOC	RPT inop TS SCRAM times	1.44	1.41	1.55	1.43
* EOC	FFTR operation Normal SCRAM times	1.33	1.32	1.40	1.34
* EOC	FFTR operation	1.40	1.38	1.50	1.40
** NA	Single Loop Operation	1.35	1.35	1.40	1.37

* Pending NRC approval of FFTR licensing submittal

** Pending NRC approval of Single Loop Licensing submittal

3.0 WNP-2 CYCLE 3 OPERATING HISTORY

WNP-2, a 3323 mwt BWR 5, began Cycle 3 operation on June 19, 1987. The end of Cycle 3 operation is expected to be April 15, 1988. During Cycle 3, 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 3 through February 12, 1988, for WNP-2. The Cycle 3 operating highlights and control rod sequence exchange schedule are found in Table 3.1.

Figure 3.1

Power History For WNP-2 For Cycle 3

TABLE 3.1

WNP-2 CYCLE 3 OPERATING HIGHLIGHTS

Began Fuel Loading	April 23, 1987
Began Commercial Operation	June 19, 1987
Projected End of Cycle Date	April 25, 1988
End of Cycle Core Average Exposure (Design)(mwd/mtm)	15,300
Number of Fresh Assemblies	148
Gross Generation (FPD) (projected)	234

Control Rod Sequence Exchange Schedule

	Sequence	
Date	From	To
September 17, 1987	A2	B2
November 10, 1987	B2	A1
January 5, 1988	A1	B1

Outages

July 6 through July 26, 1987
December 6 through 10, 1987
January 18 through January 19, 1988
February 13 through March 4, 1988*

* Estimated

4.0 RELOAD CORE DESCRIPTION

The WNP-2 core consists of 764 fuel assemblies. For the Cycle 4 reload, the core will consist of 152 ANF 8x8C fresh assemblies, 148 ANF 8x8C assemblies loaded for Cycle 3, 128 ENC 8x8C fuel assemblies loaded for Cycle 2 and 336 GE 8x8RP assemblies remaining from the initial core. The 152 ANF 8x8C fresh assemblies consist of 24 reload assemblies originally manufactured for loading in Cycle 3 and 128 reload assemblies manufactured for loading in Cycle 4. The two assemblies are identical in gadolinium oxide (GD_2O_3) loading, and in all other major physical characteristics except for enrichment. The 24 reload batch 2 assemblies have a bundle average enrichment of 2.72 weight percent U^{235} and the 128 reload batch 3 fuel assemblies have a bundle average enrichment of 2.64 weight percent U^{235}. 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 4 core.

TABLE 4.1

WNP-2 CYCLE 3 CORE

Number of Assemblies	Type	Enrichment
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
280	GE 8x8RP	2.19 w/o U-235
56	GE 8x8RP	1.76 w/o U-235

The 152 exposed GE 8x8RP assemblies discharged are all high enriched (2.19 w/o U-235) assemblies.

*Twentyfour (24) of these assemblies were originally fabricated for reload in Cycle 3 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 and have an enrichment of 2.64 weight percent U^{235}. Two of these assemblies are Lead Test Assemblies (LTA).

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

***Two of these assemblies are Lead Test Assemblies (LTA).

5.0 FUEL MECHANICAL DESIGN

The mechanical design of the 8x8C Cycle 4 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 4 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 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 4 ANF reload fuel is the same as the structural response of the 8x8C Cycle 3 ANF fuel, the 8x8C Cycle 2 ENC fuel and the 8x8RP GE fuel which also reside in the WNP-2 core. As a part of Cycle 4 operation, some of the 8x8C Cycle 4 ANF reload fuel assemblies may be channeled with new 100 mil channels fabricated by ASEA Atom as was the case for Cycle 3. 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 4 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:

- o Hydraulic compatability.
- o Fuel cladding integrity safety limit.
- o Fuel centerline temperature.
- o Bypass flow characteristics.
- o 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 Cycle 4 is 1.06 which is equal to the Cycle 1, Cycle 2, and Cycle 3 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 4 MCPR safety limit analysis methodology and input parameters are described in XN-NF-88-01, Cycle 4 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 4 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 and Cycle 3 value. 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 0.75 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 (46 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). This analysis results in a requirement to slightly modify Technical Specification 3/4.3.10.

7.0 NUCLEAR DESIGN

The neutronic methods for the design and analysis of the WNP-2 Cycle 4 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 reload batch 3 ANF reload bundles (labeled AN-3) are similar to the XN-2 ANF reload bundles in nuclear design in all major parameters except for fuel enrichment. Major nuclear design characteristics for the ANF 8x8C reload fuel assembly (AN-3) are:

- o The fuel assembly contains 62 fuel rods and two water rods. One of the water rods also acts as a spacer capture rod.
- o The fuel assembly average enrichment is 2.64 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 2.81 w/o U-235.
- o 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.
- o Each fuel assembly contains five 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 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 1.94 w/o U-235, nine rods to 2.50 w/o U-235, 16 rods to 2.86 w/o U-235, 22 rods to 3.43 w/o U-235, and five rods to 2.50 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 3 (EOC3), the core exposure for the beginning of Cycle 4 (BOC4), and the core exposure for the end of Cycle 4 (EOC4) 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 EOC3 (mwd/mtm)	15,300
Core Exposures at BOC4 (mwd/mtm)	11,200
Core Exposures at EOC4 (mwd/mtm)	16,900
BOC Cold K_{eff}, all rods out	1.1194
BOC Cold K_{eff}, strongest rod out	0.9894
Reactivity Defect/R-Value, percent % K/K	0.0
Standby Liquid Control System (SBLC) Reactivity, 660 PPM Boron, K_{eff}	0.9654

7.3 Comparison of Major Core Parameters

Some of the major core parameters for WNP-2 Cycle 3 and Cycle 4 are listed in Table 7.2.

Table 7.2

COMPARISON OF MAJOR CORE PARAMETERS

Parameter	Cycle 3	Cycle 4
MCPR Limit (0 mwd/mtm)	1.29	1.27
Doppler Defect (%K /K %T)	$- 9.5 \times 10^{-6}$	$- 9.5 \times 10^{-6}$
Cycle Length (Design; FPD)	227	227
Core Average Exposure (BOC; mwd/mtm)	9,639	11,200
Core Average Exposure (EOC; mwd/mtm)	15,300	16,900

The differences between the Cycle 3 core and the Cycle 4 core are found in the core loading pattern. The Cycle 3 core consisted 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. The Cycle 4 core will consist of a scatter load of 152 ANF 8x8 unirradiated assemblies, 148 once irradiated ANF 8x8 assemblies, 128 twice irradiated 8x8 assemblies, and 336 thrice irradiated P8x8R assemblies fabricated by General Electric (GE).

8.0 ANTICIPATED OPERATIONAL OCCURRENCES

ANF considers eight categories of potential system core wide transient occurrences for jet pump BWRs in Reference 11.0. ANF has provided analysis results for the three most limiting transients for WNP-2 Cycle 4 to determine the Cycle 4 thermal margins. The three transients determined to be most limiting for Cycle 4 are:

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

ANF's methodology for developing thermal limits is found in Reference 12.0. 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 results of the core-wide and local transient analyses are provided in the WNP-2 Cycle 4 Reload Analysis Report (Reference 1.0) and in the WNP-2 Cycle 4 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 0.17 for the ANF fuel and 0.21 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 1.27 for the GE fuel in Cycle 4 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 1.30 and for GE fuel is 1.31 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 LRNB transient and the MCPR operating limit within EOC-2000 mwd/mtm to EOC is 1.36 for ANF fuel and 1.38 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 LRNB transient and the MCPR operating limit is 1.35 for ANF fuel and 1.38 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 LRNB transient at 106 percent flow is bounding and the MCPR operating limit is 1.41 for ANF fuel and 1.44 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 0.18 for ANF fuel and, 0.22 for GE fuel at 107 percent, and 0.20 for ANF fuel, and 0.23 for GE fuel at a 108 percent rod block setting. Therefore, operation with a 108 percent RBM setting would require a MCPR limit of 1.26 for ANF and 1.29 for the GE fuel.

8.1 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 dominance of the LRNB transient over the other transients analyzed relates to a change in modeling of the WNP-2 control system for Cycle 4.

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 Δ CPR of 0.09). The WNP-2 MCPR safety limit for Cycle 4 continues to be 1.06; hence the LOFH transient requires a MCPR operating limit of 1.15 for WNP-2 (Reference 2.0).

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 0.17 for ANF fuel and 0.21 for GE fuel, for a 107 percent RBM setpoint 0.18 for ANF fuel and 0.22 for GE fuel, and for a 108 percent RBM setpoint 0.20 for ANF fuel, and 0.23 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 4. Thus, the reduced flow MCPR operating limit for WNP-2 Cycle 2 is applicable to Cycle 4.

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 11.0). The WNP-2 Cycle 4 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 well below the 110 percent design criterion.

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 dp (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.

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 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. The results of all these analyses when considered along with the similarity between the two fuel types utilized in Cycle 4, confirm the capability of WNP-2 to operate at 100 percent power and 106 percent core flow during Cycle 4 operation (Reference 16.0).

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 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 4.

8.6 Single Loop Operation

ANF recently performed analyses for WNP-2 which demonstrate the safety of plant operation with a single recirculation loop out of service 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.

- 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.

- 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 4 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 4 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.

8.7 Final Feedwater Temperature Reduction

Reference 20.0 presents a final feedwater temperature reduction (FFTR) analysis with thermal coastdown for WNP-2. 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 Δ CPR changes for the LRNB and FWCF transients of + 0.02 and - 0.01 are applicable to these respective anticipated operational occurrence (AOO) events. That is, these LRNB and FWCF limit changes are applicable when Cycle 4 reactor operation is being extended with thermal coastdown at FFTR conditions.

9.0 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 Reference 21.0. These results are equally applicable to Cycle 4. 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 4.

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 4 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 4, the analysis shows a value of 149 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

ANF recommends the use of limits ANF 8 x 8 Fuel Two Loop MPALHGR limits 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 consistent with the flow dependent MCPR curve (1.35 at 50 percent of rated flow). 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 accep- tance criteria of 10CFR50.46 for loss-of-coolant accident breaks up to and including the double-ended severance of a reactor coolant pipe.

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 4. 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.

10.1 Core Load Verification Test

Goal - To assure that the WNP-2 Cycle 4 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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