June 2, 1987

Docket No. 50-397

Mr. G. C. Sorensen, Manager
Regulatory Programs
Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way
Richland, Washington 99352

Dear Mr. Sorensen:

Subject: ISSUANCE OF AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC 64987)

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 45 to Facility Operating License NPF-21 to the Washington Public Power Supply System for WPPSS Nuclear Project No. 2, located in Benton County near Richland, Washington. This amendment is in response to your letter dated March 27, 1987, as supplemented by your letter of April 22, 1987.

This amendment revises the WNP-2 Technical Specifications as follows:

1. Sections B 2.0, B 2.1.2, 3/4.2.1, 3/4.2.4, B 3/4.1.3, B 3/4.2.1, B 3/4.2.3, and B 3/4.7.9 are revised to reflect the replacement of approximately 148 initial core fuel assemblies with Advanced Nuclear Fuels (ANF) Corporation fuel assemblies for the upcoming Cycle 3 operation.

2. Sections 3/4.1.3.4, 3/4.2.3, and 3/4.3.10, are revised to establish operating limits for Cycle 3 operation.

A copy of the related safety evaluation supporting Amendment No. 45 to Facility Operating License No. NPF-21 is enclosed.

Sincerely,

Original signed by

Robert B. Samworth, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III/IV/V
& Special Projects

Enclosures:

1. Amendment No. 45 to Facility Operating License No. NPF-21

2. Safety Evaluation

cc: w/enclosures
See next page

SEE PREVIOUS CONCURRENCES.

DRSP/PDV	DRSP/PDV	*OGC-Bethesda	DRSP/D:PDV
JLee	*RSamworth:cd		GWKnighton
6/ /87	5/15 /87	5/21/87	6/2/87

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REFERENCES

1. Letter from G. C. Sorenson, Washington Public Power Supply System (WPPSS), to U. S. Nuclear Regulatory Commission dated March 27, 1987, Nuclear Plant No. 2, Operating License NPF-21, Request for Amendment to Technical Specifications - Reload License Amendment (Cycle 3)".

2. "WNP-2 Cycle 3 Reload Summary Report" and attachment "Technical Specification Changes", WPPSS - EANF -109, March 1987.

3. "WNP-2 Cycle 3 Reload Analysis", XN-NF-87-25, March 1987.

4. "WNP-2 Cycle 3 Plant Transient Analysis", XN-NF-87-24, March 1987.

5. Letter from G. C. Sorenson, WPPSS, to U. S. Nuclear Regulatory Commission, dated April 22, 1987 "Clarification".

6. "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-86-67(P)(A), Revision 1, "September 1986.

7. Safety Evaluation for Amendment No. 28 to Facility Operating License NPF-21 to the Washington Public Power Supply System Nuclear Project No. 2, dated May 23, 1986.

8. "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads", Exxon Nuclear Company, XN-NF-80-19(A), Volume 4, Revision 1, September, 1983.

9. "Exxon Nuclear Critical Power Methodology for BWRs", XN-NF-524(P)(A), Revision 1, November 1983.

10. Letter, H. Bernard (NRC) to G. F. Owsley (ENC), "Acceptance for Referencing of Topical Report XN-NF-512, Revision 1," dated July 22, 1982.

11. General Electric Standard Application for Reactor Fuel (Supplement for United States), NEDE-24011-P-A-8-US, May 1986, with staff SER attached.

12. "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis", February 1987 (published as Advanced Nuclear Fuels report), XN-NF-105(P)(A) Vol 1, Sup 1 and Vol 1, Sup 2.

Mr. G. C. Sorensen, Manager
Washington Public Power Supply System

WPPSS Nuclear Project No. 2
(WNP-2)

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

WPPSS NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. NPF-21

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Washington Public Power Supply System (the Supply System, also the licensee), dated March 27, 1987 and clarified by subsequent letter dated April 22, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-21 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 45, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III/IV/V
& Special Projects

Enclosure: Changes to the Technical Specifications

Date of Issuance: June 2, 1987

June 2, 1987

ENCLOSURE TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following page of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
B 2-1	B 2-1
B 2-2	B 2-2
3/4 1-8	3/4 1-8
3/4 2-1	3/4 2-1
3/4 2-4	3/4 2-4
3/4 2-4C	3/4 2-4C
3/4 2-7	3/4 2-7
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 3-102	3/4 3-102
3/4 3-103	3/4 3-103
3/4 3-104	3/4 3-104
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-3	B 3/4 2-3
B 3/4 7-4	B 3/4 7-4

2.0 SAFETY LIMITS and LIMITING SAFETY SYSTEM SETTINGS

BASES

INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation for both GE and Advanced Nuclear Fuels Corporation (ANF) fuel. MCPR greater than 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding integrity Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity safety limit assures that during normal operation and during anticipated operational occurrences, at least 99.9 percent of the fuel rods in the core do not experience transition boiling (Reference XN-NF-524 (A), Rev. 1).

2.1 SAFETY LIMITS

2.1.1. THERMAL POWER, Low Pressure or Low Flow

For certain conditions of pressure and flow, the XN-3 correlation is not valid for all critical power calculations. The XN-3 correlation is not valid for bundle mass velocities less than .25 x 10^6 lbs/hr-ft^2 or pressures less than 585 psig. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/h (approximately a mass velocity of .25 x 10^6 lbs/hr-ft^2), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/h. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power

THERMAL POWER, Low Pressure or Low Flow (Continued)

at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 585 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors[a] which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the ANF nuclear critical heat fluxenthalpy XN-3 correlation. The XN-3 correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2.

The bases for the uncertainties in the core parameters are given in XN-NF-524(A), Rev. 1[a] and the basis for the uncertainty in the XN-3 correlation is given in XN-FN-512(A), Rev. 1[b]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

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a. Exxon Nuclear Critical Power Methodology for Boiling Water Reactors, XN-NF-524(A), Rev. 1.

b. Exxon Nuclear Company XN-3 Critical Power Correlation, XN-NF-512(a), Rev. 1.

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time of all operable control rods from the fully withdrawn position, for the four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.455
39	0.920
25	2.052
5	3.706

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the average scram insertion times of control rods exceeding the above limits:

1. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE for GE fuel and average bundle exposure for ANF fuel shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 when in two loop operation, and Figures 3.2.1-4, 3.2.1-5, and 3.2.1-6 when in single loop operation.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, or 3.2.1-3 in two loop operation or Figure 3.2.1-4, 3.2.1-5, or 3.2.1-6 in single loop operation, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

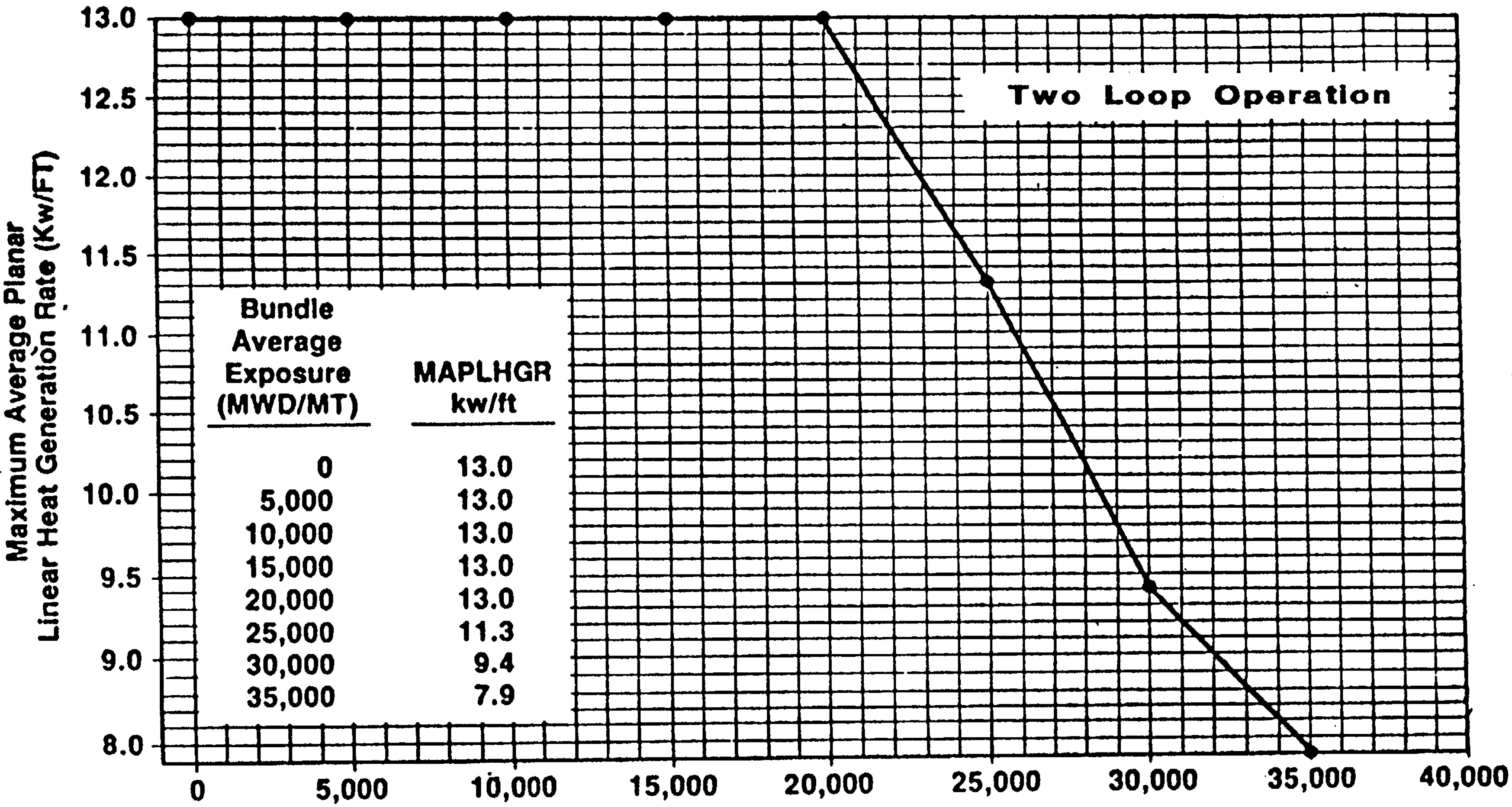
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.



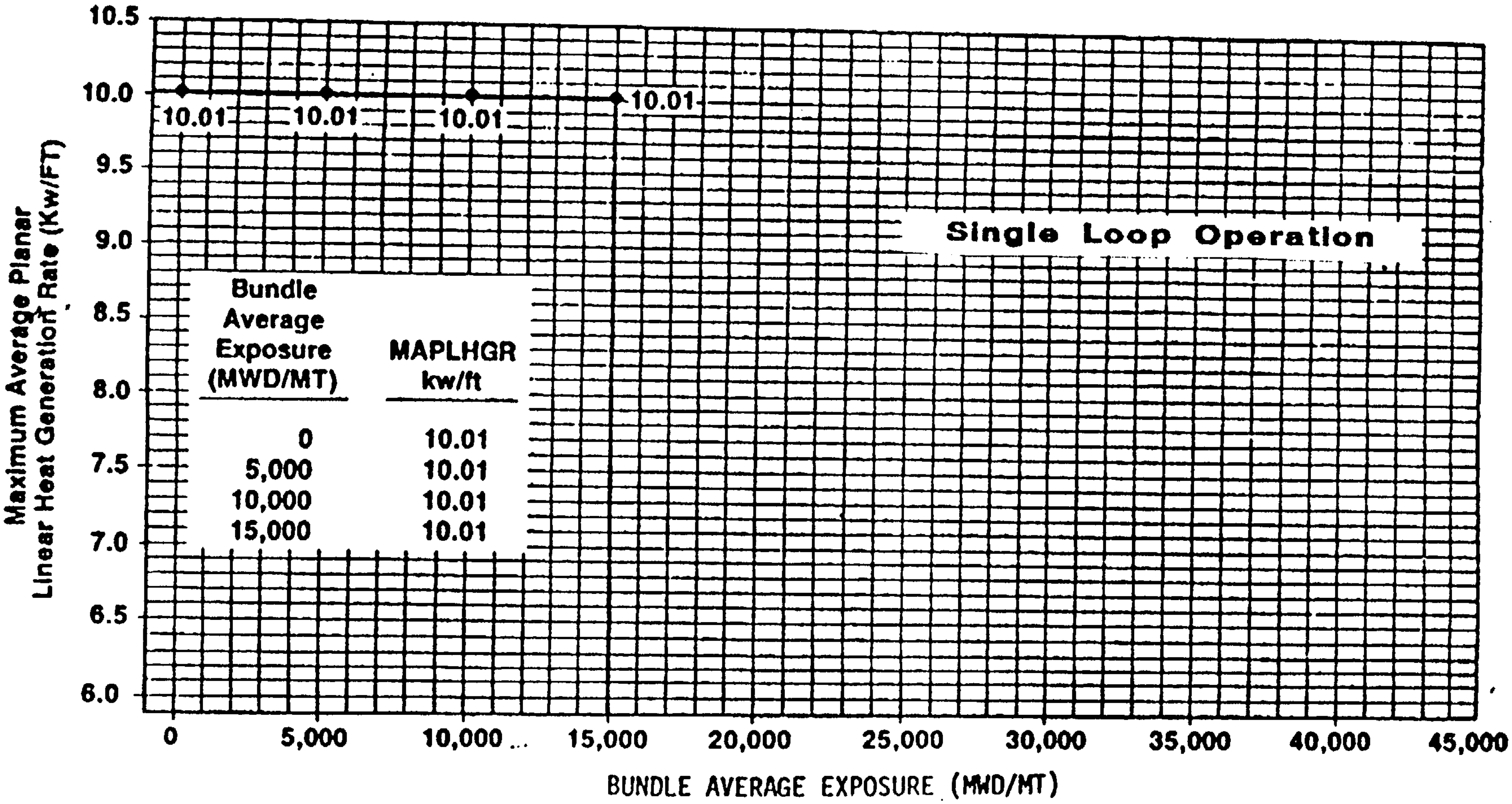
Bundle Average Exposure (MWD/MT)	MAPLHGR kw/ft
0	13.0
5,000	13.0
10,000	13.0
15,000	13.0
20,000	13.0
25,000	11.3
30,000	9.4
35,000	7.9

Bundle Average Exposure (MWD/MT)

ANF 8x8 Reload Fuel

Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Bundle Average Exposure

Figure 3.2.1-3



Bundle Average Exposure (MWD/MT)	MAPLHGR kw/ft
0	10.01
5,000	10.01
10,000	10.01
15,000	10.01

Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
BUNDLE AVERAGE EXPOSURE
ANF 8x8 Reload Fuel
Figure 3.2.1-6

Table 3.2.3-1

MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow	
			GE Fuel	ANF Fuel
1.	0 MWD/MTU - 4150 MWD/MTU	*	1.29	1.26
2.	4150 MWD/MTU - EOC MWD/MTU	Normal scram times**	1.32	1.30
3.	4150 MWD/MTU - EOC MWD/MTU	Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-7)	1.39	1.35
4.	4150 MWD/MTU - EOC MWD/MTU	RPT inoperable Normal scram times	1.37	1.35
5.	4150 MWD/MTU - EOC MWD/MTU	RPT inoperable Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-7)	1.43	1.39

*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.

**These MCPR values are based on the ANF Reload Safety Analysis performed using the control rod insertion times shown below (defined as normal scram). In the event that surveillance 4.1.3.2 shows these scram insertion times have been exceeded, the plant thermal limits associated with normal scram times default to the values associated with Tech. Spec. scram times (3.1.3.4-p 3/4 1-7), and the scram insertion times must meet the requirements of Tech. Spec. 3.1.3.4.

Position Inserted From Fully Withdrawn	Slowest measured average control rod insertion times to specified notches for all operable control rods for each group of 4 control rods arranged in a a two-by-two array (seconds)
Notch 45	.404
Notch 39	.660
Notch 25	1.504
Notch 5	2.624

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) for GE fuel shall not exceed 13.4 kW/ft. The LHGR for ANF fuel shall not exceed the values shown in Figure 3.2.4-1.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

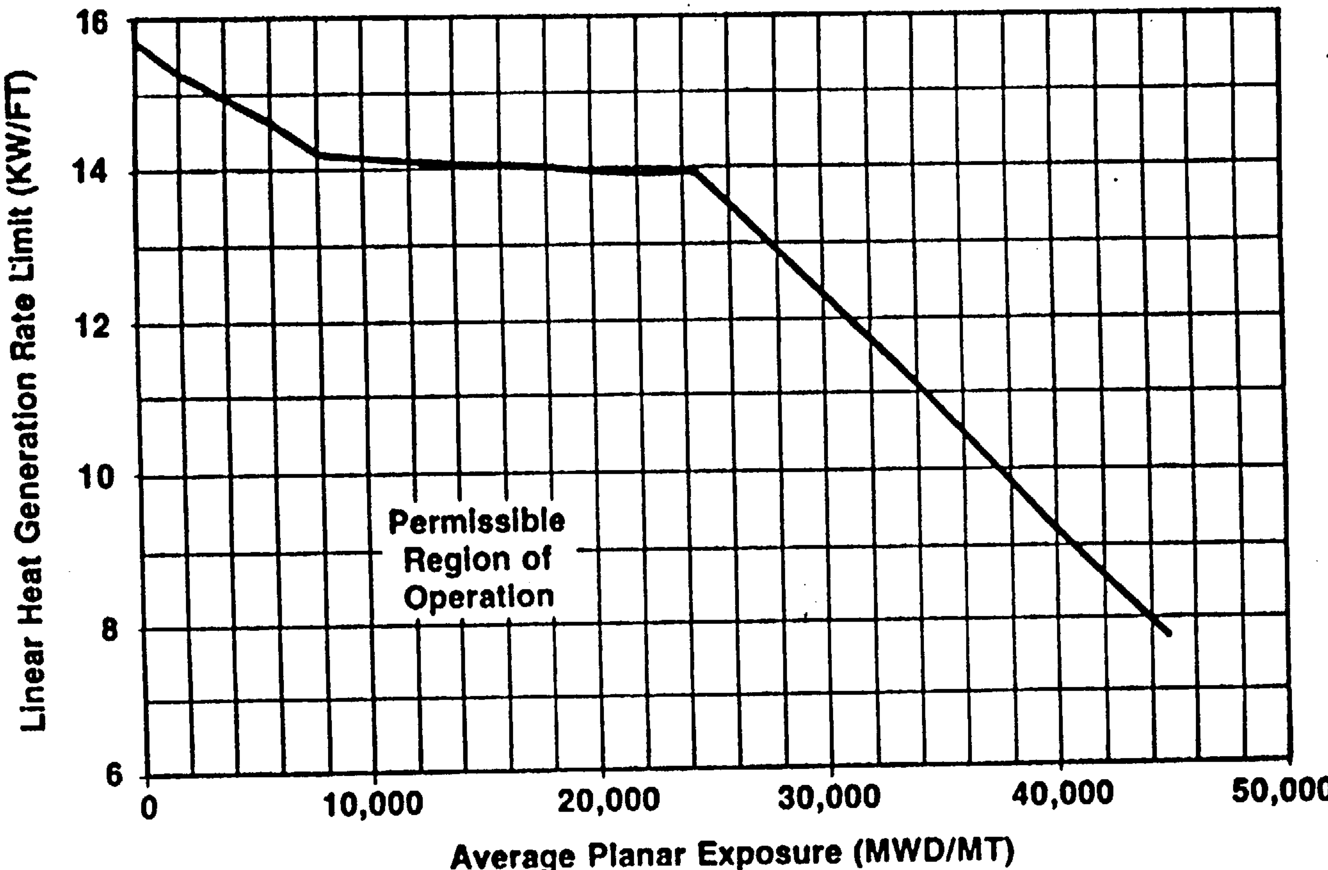
SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.



EXP	LHGR
0	15.62
510	15.621
2,580	15.10
5,230	14.71
7,940	14.19
10,470	14.13
13,220	14.06
15,990	14.06
18,708	14.00
21,590	13.93
24,420	13.93
27,280	13.08
30,150	12.24
33,050	11.40
35,960	10.47
38,900	9.55
41,830	8.65
44,760	7.77

ANF 8x8 Reload Fuel
Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure

Figure 3.2.4-1

3/4.3.10 NEUTRON FLUX MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.10 The APRM and LPRM* neutron flux noise levels shall not exceed three (3) times their established baseline values when operating in the allowable region of Figure 3.3.10-1.

APPLICABILITY: OPERATIONAL CONDITION 1 with two reactor coolant system recirculation loops in operation with THERMAL POWER greater than the limit specified in Figure 3.3.10-1 and total core flow less than 45% of rated total core flow or with one reactor coolant system recirculation loop not in operation with THERMAL POWER greater than the limit specified in Figure 3.3.10-1.

ACTION:

a. With the APRM or LPRM* neutron flux noise level greater than three (3) times their established baseline noise levels, initiate corrective action within 15 minutes to restore the noise levels to within the required limits within 2 hours or reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.3.10-1 within the next 2 hours.

b. With reactor power/core flow in the crosshatched region of Figure 3.3.10-1, initiate corrective action within 15 minutes to reduce power by control rod insertion to a reactor power/core flow below the crosshatched region within 2 hours.

SURVEILLANCE REQUIREMENTS

4.3.10.1 The provisions of Specification 4.0.4 are not applicable.

4.3.10.2 With two reactor coolant system recirculation loops in operation, establish a baseline APRM and LPRM* neutron flux noise level value within 2 hours upon entering the APPLICABLE OPERATIONAL CONDITION of Specification 3.3.10 provided that baselining has not been performed since the most recent CORE ALTERATION.

4.3.10.3 With one reactor coolant system recirculation loop not in operation, establish a baseline APRM and LPRM* neutron flux noise level value with THERMAL POWER less than or equal to the limit specified in Figure 3.3.10-1 prior to entering the APPLICABLE OPERATIONAL CONDITION of Specification 3.3.10 provided baselining has not been performed with one reactor coolant system recirculation loop not in operation since the most recent CORE ALTERATION.#

INSTRUMENTATION

NEUTRON FLUX MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS (Continued)

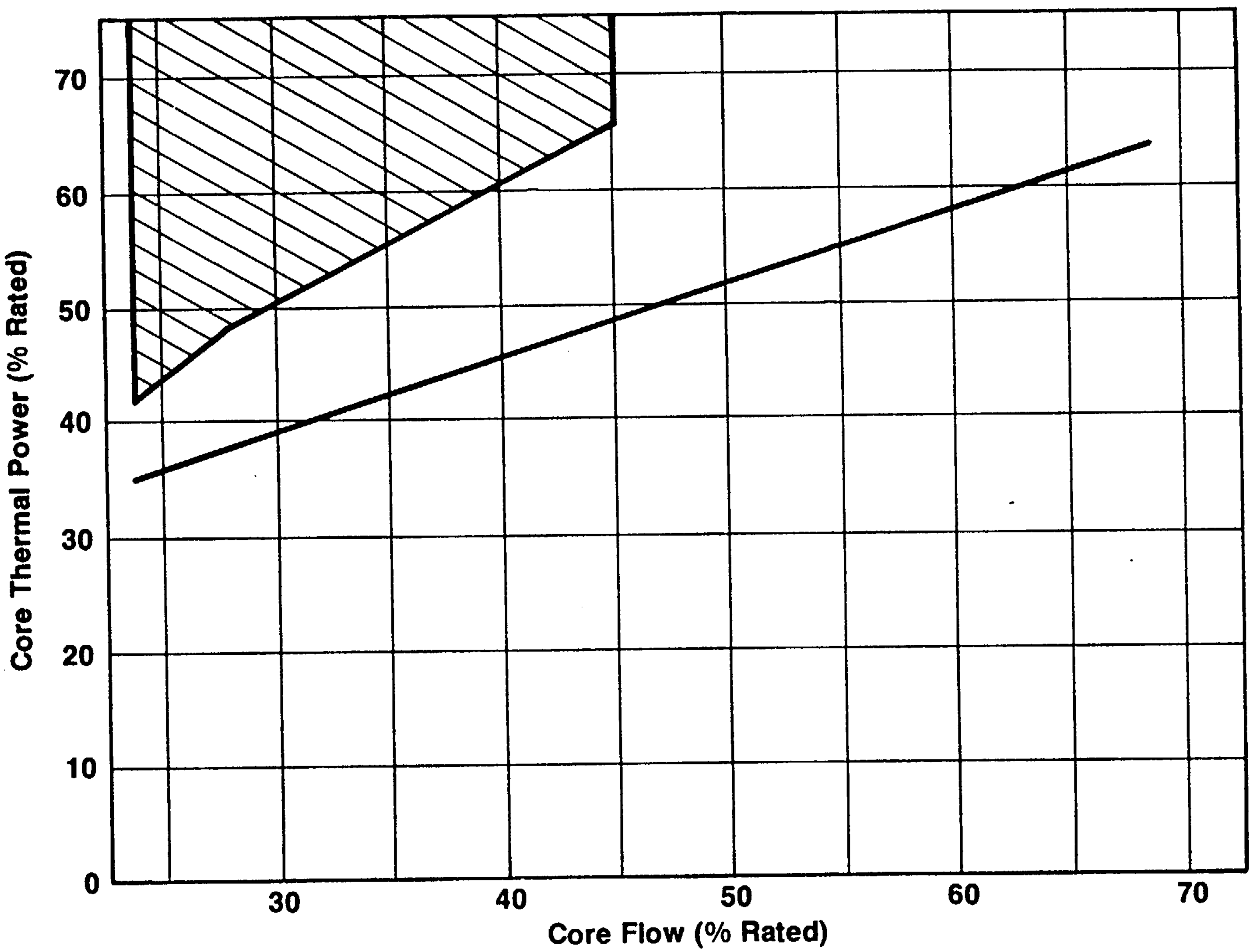
4.3.10.4 The APRM and LPRM* neutron flux noise levels shall be determined to be less than or equal to the limit of Specification 3.3.10 and the reactor power/core flow shall be verified to lie outside the crosshatched region of Figure 3.3.10-1 when operating within the APPLICABLE OPERATIONAL CONDITION of Specification 3.3.10:

a. At least once per 8 hours, and

b. Within 30 minutes after completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

*Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center of the core should be monitored.

#The baseline data obtained in Specification 4.3.10.3 is applicable to operation with one reactor coolant system recirculation loop not in operation and THERMAL POWER greater than the limits specified in Figure 3.3.10-1.



Thermal Power Limits of Specification 3.3.10-1
Figure 3.3.10-1

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the safety analyses, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the fuel cladding safety limit during the core wide transient analyzed in the cycle specific transient analysis report. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the fuel cladding safety limit. The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. For GE fuel, the peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady-state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for GE fuel is this LHGR of the highest powered rod divided by its local peaking factor which results in a calculated LOCA PCT much less than 2200°F. The Technical Specification APLHGR for ANF fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 for two recirculation loop operation. These values shall be multiplied by a factor of 0.84 for single recirculation loop operation. This multiplier is determined from comparison of the limiting analysis between two recirculation loop and single recirculation loop operation.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. These models are described in Reference 1 or XN-NF-80-19, Volumes 2, 2A, 2B and 2C, Rev. 1.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Table 3.2.3-1.

The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to an ANF core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and nonpressurization events are described in XN-NF-79-71(P) and XN-NF-84-105(A). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$ of Figure 3.2.3-1 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the maximum of the rated flow MCPR determined from Table 3.2.3-1 and the reduced flow MCPR determined from Figure 3.2.3-1, $MCPR_f$ assures that the Safety Limit MCPR will not be violated. $MCPR_f$ is only calculated for the manual flow control mode. Automatic flow control operation is not permitted.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurances that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the weight and pressure of the tanks.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.7 FIRE-RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.8 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY.

3/4.7.9 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the feedwater controller failure analysis of the cycle specific analysis. The main turbine bypass system provides pressure relief during the feedwater controller failure event so that the safety limit MCPR is not violated.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

WPPSS NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter from Mr. G. Sorensen, Washington Public Power Supply System, to the Nuclear Regulatory Commission (Ref. 1), technical specification changes were proposed for the operation of Washington Nuclear Plant No. 2 (WNP-2) for Cycle 3 (N2C3) with a fuel reload using Advanced Nuclear Fuels (ANF) Corporation fuel assemblies and ANF analyses and methodologies. Enclosed were the proposed Technical Specification changes and several reports (References 2-4) discussing the reload and analyses which support and justify the third cycle operation with General Electric (GE) and ANF fuel and the proposed technical specification changes.

A subsequent letter (Ref. 5) was submitted which provided a discussion and results of a plant-specific Loss of Feedwater Heating Transient analysis and other information requested by the NRC. Cycle 3 is the second use of the ANF (previously Exxon Nuclear Company) 8x8C fuel assemblies and analytical methodologies for this reactor. Similar reloads with the ANF fuel type have been done for Dresden Units 2 and 3 and Susquehanna Steam Electric Station, Units 1 and 2. These reloads and the associated ANF methodologies have been extensively reviewed and approved by the NRC staff and are generally applicable for N2C3 analyses.

2.0 EVALUATION

2.1 Reload Description

The N2C3 reload will retain 488 General Electric (GE) and 128 Exxon Nuclear Company (ENC) XN-1 fuel assemblies from the previous cycle and will add 148 ANF manufactured 8x8C, 2.72 percent average, 2.89 percent peak radial average U235 enriched fuel assemblies. The ANF 8x8C fuel assemblies are essentially the same as the ENC XN-1 assemblies loaded in the previous Cycle 2 reload. The loading pattern will be a conventional scatter pattern with low reactivity fuel on the periphery.

2.2 Fuel Mechanical Design

The ANF 8x8C fuel assemblies used for N2C3 are the same as those previously designated XN-1 and have been approved generically by the NRC staff for ANF reload cores (Ref. 6). There are slight differences in the fuel and plug designs, but the enrichment, gadolinium placement and significant mechanical and thermal-hydraulic design elements are the same and the power distributions are similar. The methodologies used for the fuel design and analysis are the same as those used for the prior Cycle 2 reload for this facility (see Ref. 7 for detailed discussion). The design and analyses of the ANF fuel assembly as used in N2C3 are thus acceptable.

Some of the N2C3 8x8C reload fuel assemblies will be channeled with new 100 mil channels fabricated by ASEA-ATOM. Based on our review of additional information provided by the licensee in Attachment 4 to Reference 5 relative to physical data for the new channels and the previous use of the channels in other facilities, the staff concludes that the performance of the ASEA-ATOM channels will be the same as the original GE channels and that the use of the new channels is acceptable.

2.3 Nuclear Design

The nuclear design for N2C3 has been performed with ANF methodologies previously reviewed and approved, and were used in the Cycle 2 (N2C2) analysis. The overall methodology is described in the ENC (now ANF) Licensing Topical Report XN-NF-80-19(A), Volume 4, Revision 1 (Ref. 8). The fuel loading pattern is given in Figure 4.2 of Reference 3. The shutdown margin (SDM) at the beginning of the cycle and at minimum conditions is 1.18 percent delta k, well in excess of the required 0.38 percent delta k. The Standby Liquid Control System also fully meets shutdown requirements. Since these results have been obtained by the use of previously approved methods and fall within the expected range, we conclude that the nuclear design of the N2C3 reload core is acceptable.

2.4 Thermal Hydraulic Design

The ANF thermal hydraulic methodology and criteria used for the N2C3 design and analysis is the same as the prior N2C2 reload. The previous review concluded that hydraulic compatibility between GE and ANF fuel is satisfactory and the calculation of core bypass flow and the Safety Limit Minimum Critical Power Ratio (SLMCPR) are acceptable. The methodology for Cycle 3 is based on ANF's revised critical power methodology in XN-NF-524, Revision 1 (Ref. 9) which incorporates a constant flow MCPR for formulation for BWR applications. The staff has completed its generic review of XN-NF-524 and has concluded that the methodology for arriving at a SLMCPR is acceptable. The XN-3 correlation used to develop the SLMCPR has been approved for application to both the ANF 8x8C and GE 8x8R fuel types (Ref. 10). The staff approval of References 9 and 10 includes approval of the values for generic nuclear uncertainties. The staff questioned the licensee's value of 1.06 for the SLMCPR as applied to GE 8x8R reload fuel types, since the value represents a decrease in thermal margin over that specified in GESTAR II, Section S.2.1 (Ref. 11). In response to staff questions, the licensee provided additional discussion (Attachment 3 to Ref. 5) to justify the decrease in thermal margin over that specified in GESTAR-II which would be required for reloads involving only GE fuel types. Based on the staff review of the additional material provided by the licensee, which describes reduced uncertainties in the TIP readings associated with the use of the POWERPLEX core simulator employed in the WPPSS core power distribution technique, and the previous acceptance of the POWERPLEX system in N2C2, the staff finds the additional considerations sufficient to justify the SLMCPR of 1.06 for GE fuel and the value is acceptable for N2C3.

2.5 Transient and Accident Analyses

The ANF transient methodology is basically the same as that used and approved for the previous reload Cycle 2 (N2C2). Certain aspects of the methodology as identified in the following discussion have received more recent NRC approval.

ANF examined the standard transient events and the N2C3 Transient Analysis Report (Ref. 4) which presented the results for the more limiting events. The most limiting core wide transients were the Load Rejection Without Bypass (LRWB) and the Feedwater Controller Failure (FWCF). These events were analyzed at increased core flow (106 percent) and both normal and standard Technical Specification (TS) required scram times, and with Recirculation Pump Trip (RPT) operable and inoperable. The concept of normal and TS scram times was discussed and approved as part of the N2C2 reload review (Ref. 7). The additional aspect of the ANF plant transient model recently approved by the staff is the XCOBRA-T code (Ref. 12) which is used in the determination of the thermal margins for the transients. The analyses were all done with approved methodologies and the results are acceptable.

The original reload submittal included an analysis of the Loss of Feedwater Heating (LOFWH) transient based on a generic approach proposed by ANF. Since the staff has not approved a generic ANF analytical methodology, the licensee provided a plant-specific analysis in Attachment 1 to Reference 5 which the staff finds as an acceptable approach for Cycle 3 only.

Two local events, Control Rod Withdrawal Error (CRWE) and Fuel Loading Error (FLE), were analyzed using approved methodology with the result that the CRWE was shown to have a limiting OLMCPR for a certain portion of the fuel cycle. The values are included as part of the proposed Technical Specification changes.

Compliance with overpressurization criteria was demonstrated by analysis of Main Steam Isolation Valve (MSIV) closure with MSIV position switch failure. Six safety-relief valves were assumed out of service. Maximum pressure was 105 percent of vessel design pressure, well under the 110 percent criterion. The calculation was done with approved methodology and the results are acceptable.

The LOCA analysis for the Cycle 2 was performed for a full core of ANF 8x8C fuel and remains applicable for the Cycle 3 residual and reload ANF fuel. This LOCA analysis has covered an acceptable range of conditions, has been performed with approved methodology and the resulting technical specification MAPLHGR values for the ANF fuel remain acceptable.

The rod drop accident was analyzed with approved ANF methodology. The resulting maximum fuel enthalpy of 170 cal/gm is within the established limit of 280 cal/gm. The analysis and results are acceptable.

Our review of the transient and accident analyses done for N2C3 indicated that appropriate methodology and input have been used and the results provide a suitable basis for the proposed N2C3 technical specifications.

3.0 TECHNICAL SPECIFICATION CHANGES

The following WNP-2 Technical Specifications and Bases changes have been proposed for operation during reload Cycle 3:

(1) Bases pages B 2-1 and B 2-2, Limiting Conditions of Operation (LCO) pages 3/4 1-8 and 3/4 2-1, Figure 3.2.1-3 on page 3/4 2-4, Figure 3.2.1-6 on page 3/4 2-4C, LCO page 3/4 2-9, Figure 3.2.4-1 on page 3/4 2-10, and Bases pages B 3/4 1-2, B 3/4 2-3 and B 3/4 7-4: Changes were made to reflect the corporate change from Exxon Nuclear Company (ENC) to Advanced Nuclear Fuel (ANF) Corporation, to identify the new fuel designation from ENC XN-1 to ANF 8x8C and to incorporate editorial changes. A reference to the approved thermal margin methodology report, XN-NF-84-105(A) was added. These changes are administrative only and have no safety significance, and are, therefore, acceptable.

It is also noted, based on a statement in Reference 5, that LCO page 3/4 2-1 contains changes related to the previous Cycle 2 reload analysis which were inadvertently omitted in Amendment 28. The correct, acceptable replacement page 3/4 2-1 is provided as Attachment 2 to Reference 5 and is included in this amendment's technical specification page changes.

(2) Table 3.2.3-1, page 3/4 2-7: The previous table of MCPR operating limits for rated core flow is deleted and replaced with a table stating the approved values for Cycle 3. This change is acceptable.

(3) LCO page 3/4 - 102: An ACTION statement (b) was added which states:

With reactor power/core flow in the crosshatched region of Figure 3.3.10-1, initiate corrective action within 15 minutes to reduce power by control rod insertion to a reactor power/core flow below the crosshatched region within 2 hours."

This statement applies to the single loop mode of operation (SLO) which has been approved since the first cycle of operation of WNP-2, and serves to specify the time limits for corrective action to be consistent with the presently approved action statement of TS Section 3.3.10. The proposed statement is similar to that for SLO Technical Specifications which have been approved on other facilities by the staff and is acceptable.

(4) Section 4.3.10.4, page 3/4 3-103: A phrase was added to the surveillance requirement for SLO to require that the reactor power/core flow shall be verified to lie outside the crosshatched region of Figure 3.3.10-1. This is a companion requirement to the additional statement in item (3) above to identify the region for corrective action. The addition is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation and use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off-site, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONTACT WITH STAFF OFFICIAL

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the *Federal Register* (52 FR 13352) on April 22, 1987, and consulted with the State of Washington. No public comments were received, and the State of Washington did not have any comments.

6.0 CONCLUSIONS

We have reviewed the reports submitted for the Cycle 3 reload of WNP-2 with ANF fuel and with ANF methodology and analysis. Based on this review we conclude that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The proposed technical specification changes submitted for this reload suitably reflect the use of acceptable methodology and the operating limits associated with those changes and reload parameters.

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The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (50 FR 29021) on July 17, 1985, and consulted with the state of Washington. No public comments were received, and the state of Washington did not have any comments.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: M. McCoy

Dated:

DSRP/PD5	DSRP/PD5		DSRP/D:PD5
RSamworth	JLee		GWKnighton
5/15/87	5/ /87	5/ /87	5/ /87

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Principal Contributor: M. McCoy

Dated: June 2, 1987