June 7, 1989

Docket No. 50-397

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OGC (for info only)	Region V (4)
DHagan	

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

Dear Mr. Sorensen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 72251)

The U.S. Nuclear Regulatory Commission has issued the enclosed amendment 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 3, 1989 (G02-89-029) and supplement dated April 20, 1989 (G02-89-067) and June 1, 1989 (G02-89-102).

This amendment revises Technical Specification Sections 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," 5.3, "Reactor Core," and Bases Sections B2.1.2, "Thermal Power, High Pressure and High FLow", and B3/4.2.1, "Average Planar Linear Heat Generation Rate." These revisions provide the operating limits established for the fifth cycle of operation. They also accommodate the inclusion of four ANF 9 by 9 lead fuel assemblies in the reactor core.

A copy of the related safety evaluation supporting the amendment is enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

original signed by

Robert B. Samworth, Senior Project Manager
Project Directorate V Division of Reactor Projects - III,
IV, V and Special Projects Office of Nuclear Reactor Regulation

8906160039 890607
PDR ADOCK 05000397
P PNU

Enclosures:
1. Amendment No. 69 to Facility Operating License No. NPF-21
2. Safety Evaluation

cc: w/enclosures See next page
*See previous concurrence

DF01
1/1

DRSP/PD5*	DRSP/PD5*	OGC*	DRSP/D:PD5
JLee	RSamworth:dr	MYoung	GWKalman
5/1/89	5/16/89	5/26/89	6/7/89

CP1

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Robert B. Samworth, Senior Project Manager
Project Directorate V Division of Reactor Projects - III,
IV, V and Special Projects Office of Nuclear Reactor Regulation

Enclosures: 1. Amendment No. to Facility Operating License No. NPF-21 2. Safety Evaluation

cc: w/enclosures See next page

DRSP/PD5	DRSP/PD5	OGC w/noted revisions to SE.	DRSP/D:PD5
JLee	RSamworth:dr		GWKnighton
5/11/89	5/16/89	5/26/89	5/ /89

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

June 7, 1989

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. 69 TO FACILITY OPERATING LICENSE NO. NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 72251)

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Robert B. Samworth

Robert B. Samworth, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 69 to Facility Operating License No. NPF-21
2. Safety Evaluation

cc: w/enclosures
See next page

Mr. G. C. Sorensen

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

NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
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 licensee), dated March 3, 1989 as supplemented on April 20, 1989 and June 1, 1989 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.

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P PNU

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of 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. 69, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 7, 1989

ENCLOSURE TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Also to be replaced are the following overleaf pages.

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xx	xix
B 2-2	B 2-1
3/4 1-8	3/4 1-7
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3/4 2-4C	-
3/4 2-7	-
3/4 2-8	-
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3/4 2-10A	-
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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.

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-NF-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.

---

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.

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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	0.430
39	0.868
25	1.936
5	3.497

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, 3.2.1-3, and 3.2.1-6 when in two loop operation, and Figures 3.2.1-3, 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, 3.2.1-3 or 3.2.1-6 in two loop operation or Figure 3.2.1-3, 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.

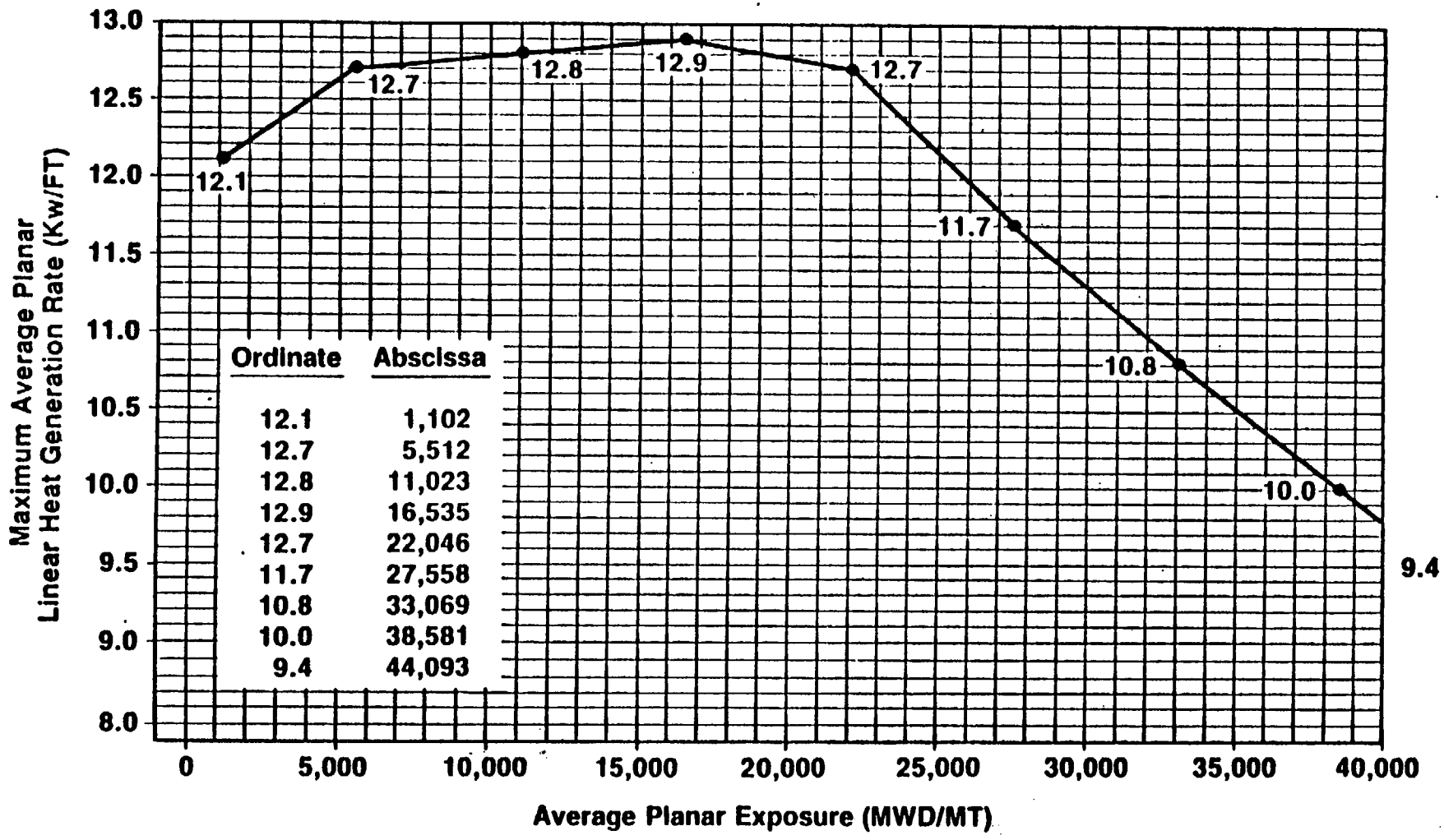
SURVEILLANCE REQUIREMENTS

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.

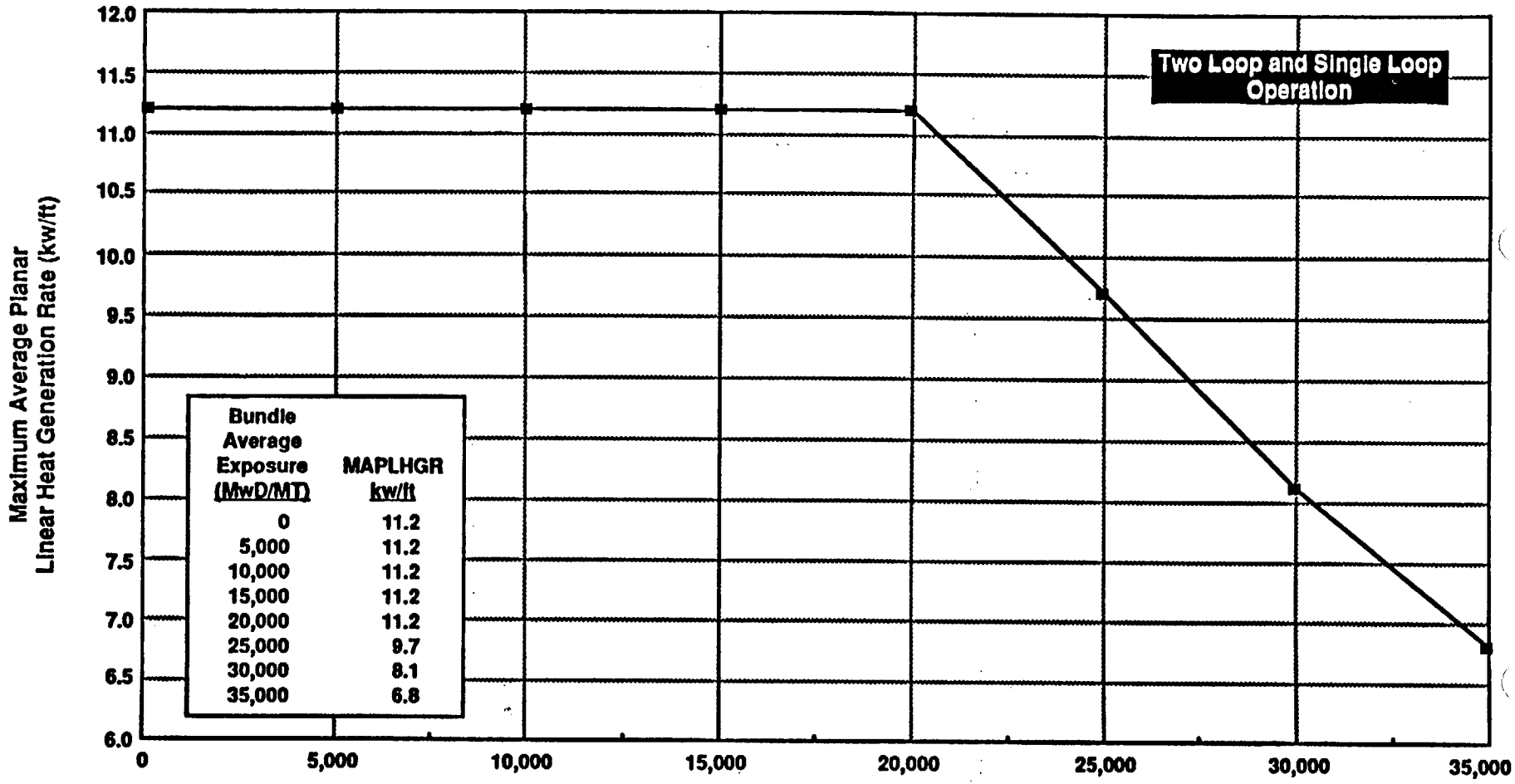


Ordinate	Abscissa
12.1	1,102
12.7	5,512
12.8	11,023
12.9	16,535
12.7	22,046
11.7	27,558
10.8	33,069
10.0	38,581
9.4	44,093

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Type 8CR183

Figure 3.2.1-1

860599.4A



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

Figure 3.2.1-6

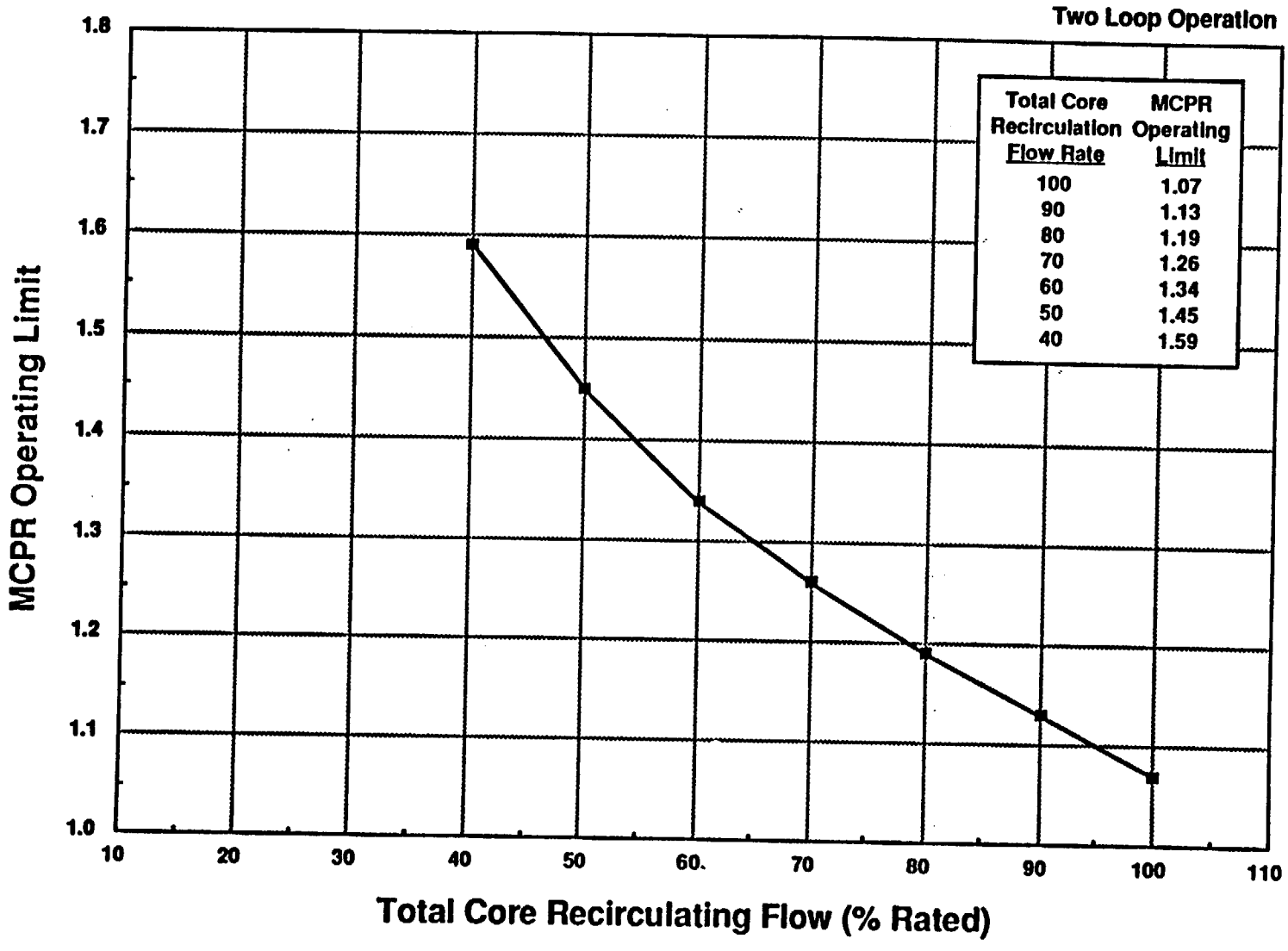
Table 3.2.3-1
MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow	
			GE Fuel	ANF Fuel
1.	0 $\frac{MWD}{MTU}$ - 3750 $\frac{MWD}{MTU}$	*	1.24	1.24
2.	3750 $\frac{MWD}{MTU}$ - EOC $\frac{MWD}{MTU}$	Normal scram times**	1.35	1.31
3.	3750 $\frac{MWD}{MTU}$ - EOC $\frac{MWD}{MTU}$	Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.42	1.38
4.	3750 $\frac{MWD}{MTU}$ - EOC $\frac{MWD}{MTU}$	RPT inoperable Normal scram times**	1.42	1.38
5.	3750 $\frac{MWD}{MTU}$ - EOC $\frac{MWD}{MTU}$	RPT inoperable Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.48	1.42
6.	0 $\frac{MWD}{MTU}$ - EOC $\frac{MWD}{MTU}$	Single loop operation RPT operable Normal scram times**	1.35	1.35

*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 1-8), 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



Note: This curve is also applicable to FFTR operation when approved

Reduced Flow MCPR Operating Limit
Figure 3.2.3-1

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 Figures 3.2.4-1, 3.2.4-2, 3.2.4-3.

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.

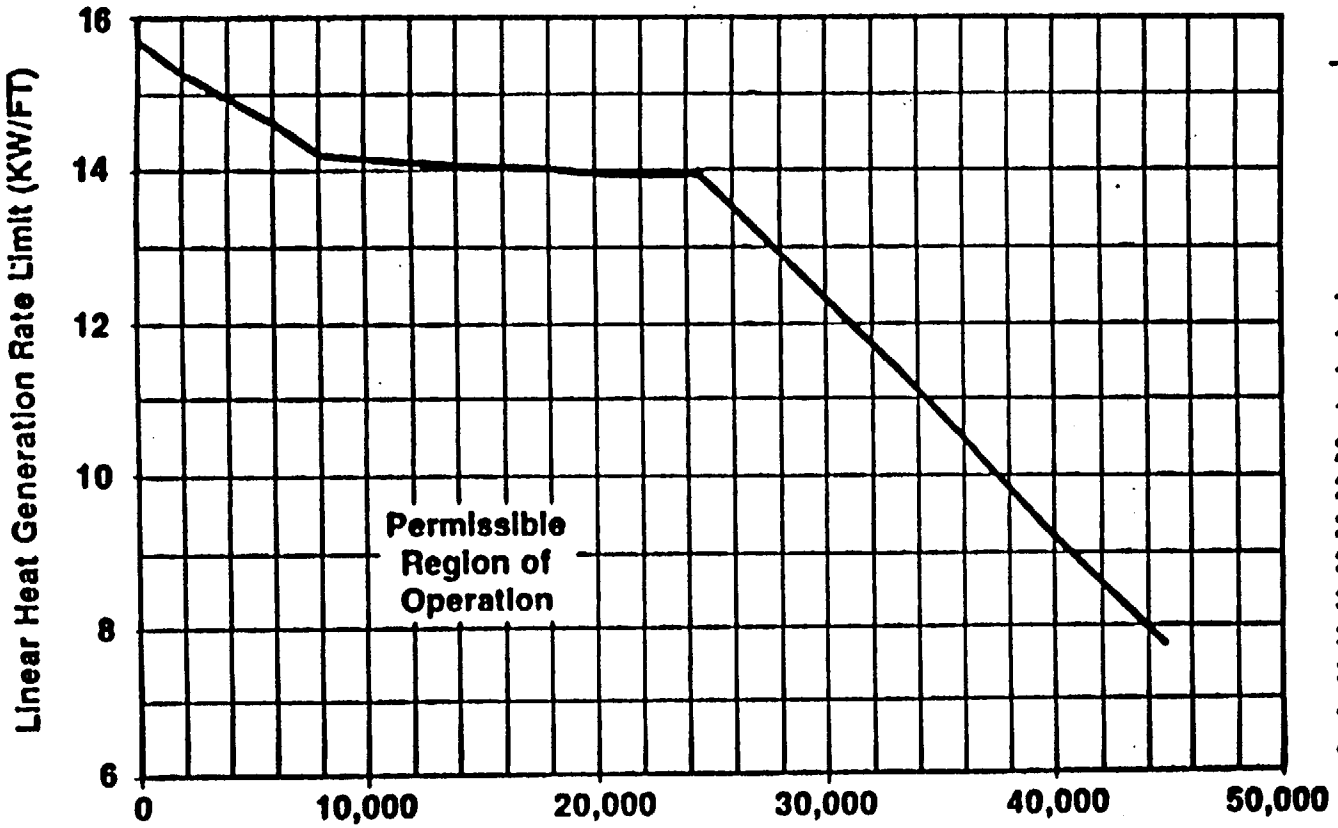
SURVEILLANCE REQUIREMENTS

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

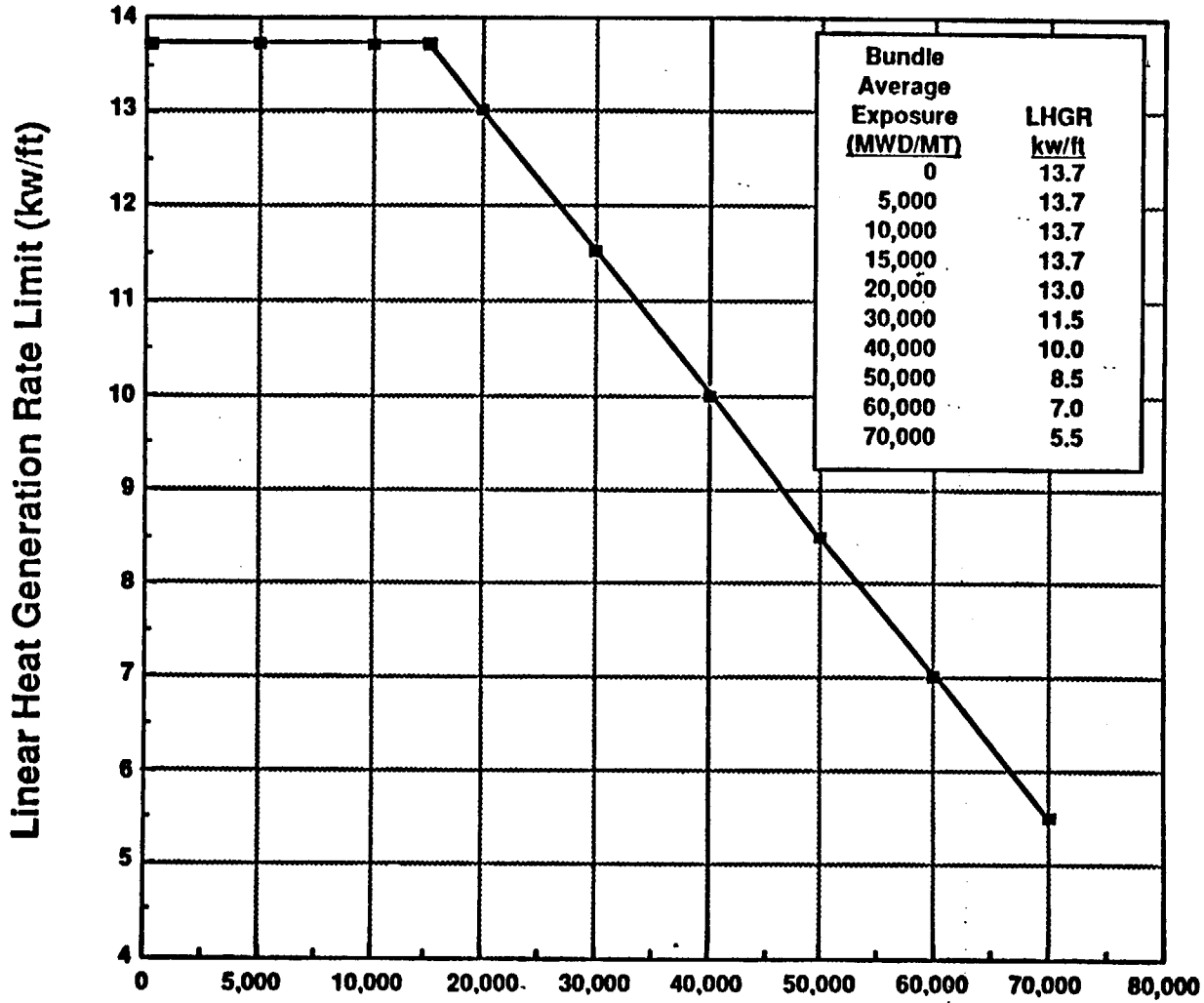
c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.



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

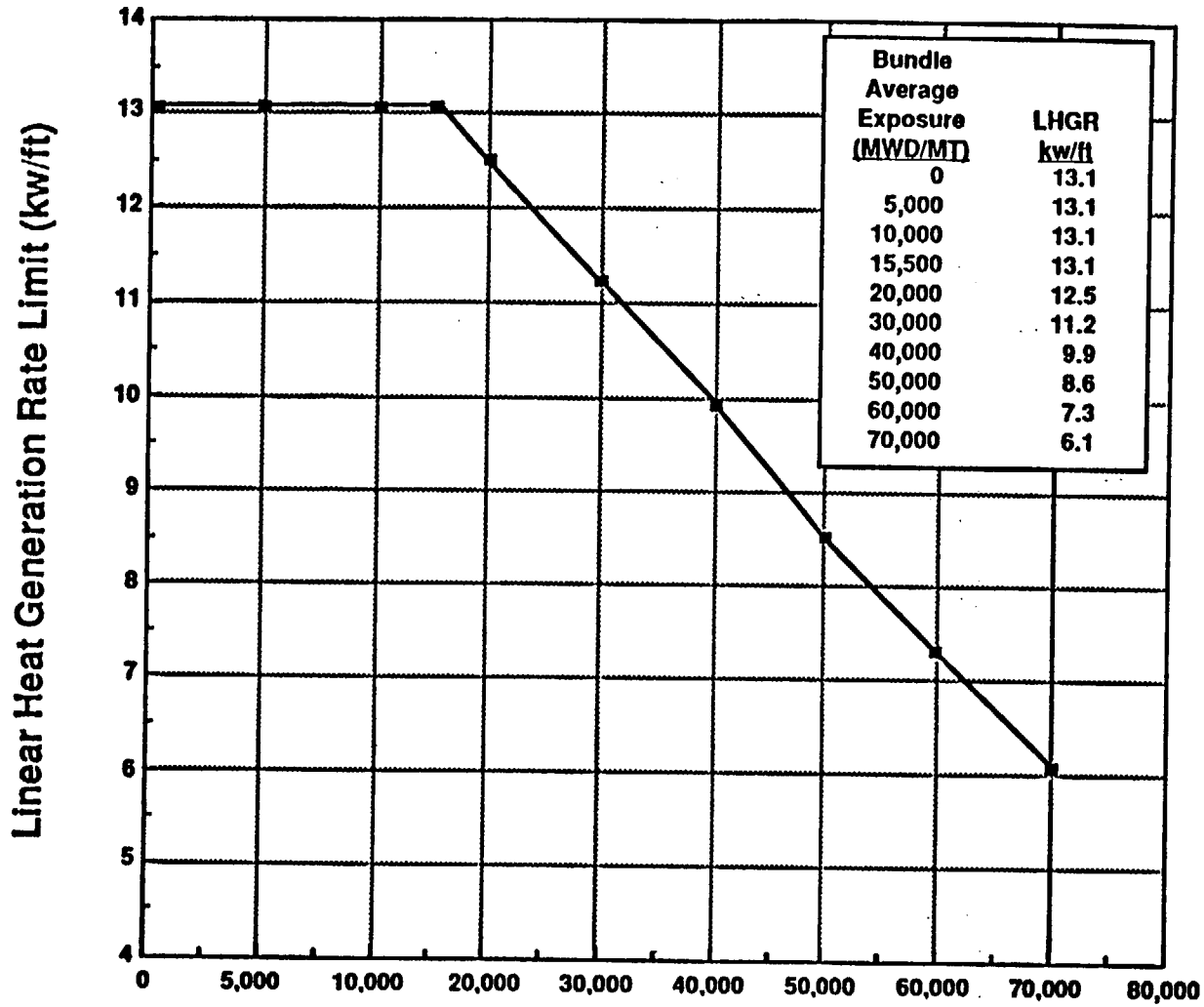
ANF 8x8 Reload Fuel
Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure

Figure 3.2.4-1



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

Figure 3.2.4-2



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

Figure 3.2.4-3

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 and 3.2.1-2 for two recirculation loop operation and Figures 3.2.1-4 and 3.2.1-5 for single loop operation. Figures 3.2.1-3, and 3.2.1-6 apply to both single and two loop operation.

The calculational procedure used to establish the APLHGR shown on 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 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 NEDC-20566P or XN-NF-80-19, Volumes 2, 2A, 2B and 2C, Rev. 1.

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analysis. In addition, the APRM setpoints must be adjusted for both two recirculation loop operation and single recirculation loop operation to ensure that the MCPR does not become less than the fuel cladding safety limit or that > 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each initial core fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy-2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.90 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading except that the reload fuel may employ a 9 x 9 array of fuel rods.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable surveillance requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.
2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,539 cubic feet at a nominal steam dome saturation temperature of 545°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal 6.5-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.95 when flooding with water is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 605 ft 7 in.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2658 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letters dated March 3, 1989, April 20, 1989 and June 1, 1989 (Ref. 1), Washington Power Supply System (WPPSS), the licensee, proposed to amend Operating License NPG-21 to support Cycle 5 operation of their Nuclear Plant No. 2 (WNP-2) with Advanced Nuclear Fuels Corporation (ANF) reload fuel. In support of the Cycle 5 reload, the licensee submitted reports consisting of a reload summary (Ref. 2), the reload analysis (Ref. 3), the plant transient analysis (Ref. 4), and the proposed Technical Specification (TS) changes.

The April 20, 1989 submittal provided additional analyses to support the proposed changes and clarifications to contain TS changes. The June 1, 1989 submittal provided graphic quality copies of the four figures affected by the proposed amendment. Additional detail was provided for one of the four figures (Figure 3.2.3-1). These submittals did not alter the action noticed in the Federal Register on April 5, 1989 and did not affect the initial no significant hazards determination.

2.0 EVALUATION

2.1 Reload Description

The WNP-2 Cycle 5 reload will incorporate a total of 136 unirradiated ANF 8x8C fuel assemblies which replace 136 of the General Electric (GE) initial core fuel assemblies. The remainder of the core is comprised of 152 ANF 8x8C assemblies loaded for Cycle 4, 148 ANF 8x8C assemblies loaded for Cycle 3, 128 ANF 8x8C assemblies loaded for Cycle 2 and 200 GE P8x8R assemblies remaining from the initial core. The licensee is also requesting approval for loading four 9x9 ANF lead fuel test assemblies (LTAs).

2.2 Fuel Design

The mechanical design of the ANF 8x8C reload fuel is described in References 6, 7 and 8. The remaining fuel types to be returned to the Cycle 5 core were approved for operation in previous cycles.

The 136 8x8C ANF reload fuel assemblies manufactured for loading in Cycle 5 are essentially identical to the 8x8C ANF reload assemblies originally

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fabricated for reload in Cycle 2 in all major physical characteristics except U-235 enrichment. All of the reload fuel assemblies are essentially identical to the 8x8C ENC fuel approved for use in Cycle 2 (Ref. 5). Based on this, and on the fuel mechanical design analysis which used approved methodologies (Ref. 8) and yielded acceptable results, the staff finds the mechanical design of the ANF 8x8C reload fuel for the WNP-2 Cycle 5 reload acceptable.

2.3 Thermal-Hydraulic Design

The ANF thermal-hydraulic methodology and criteria used for the Cycle 5 design and analysis is the same as the previous WNP-2 reloads. These previous reviews concluded that hydraulic compatibility between ANF and GE fuel is satisfactory and the calculation of core bypass flow and the safety limit minimum critical power ratio (MCPR) is acceptable. The methodology for Cycle 5 is based on ANF's revised critical power methodology (Ref. 9) which incorporates a constant flow MCPR formulation for BWR applications and has been approved by the staff. The XN-3 correlation used to develop the MCPR limit has been approved for application to both the ANF 8x8C and GE 8x8R fuel types (Ref. 10). Therefore, the proposed safety limit MCPR of 1.06 for all fuel types in this reload is acceptable.

2.4 Thermal Hydraulic Stability

The licensee, by letter dated March 3, 1989 (G02-89-030), submitted its response to IE Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors." WPPSS' letter dated March 31, 1989 (G02-89-051) also requested Technical Specification changes for power flow instability and neutron flux noise monitoring. This request is being reviewed separately and a separate document will be issued addressing the thermal hydraulic stability concerns. Completion of the thermal hydraulic review is not a prerequisite for the Cycle 5 reload.

2.5 Nuclear Design

The nuclear design for Cycle 5 has been performed with ANF methodologies previously reviewed and approved (Ref. 11). The fuel loading pattern is given in Figure 4.2 of Reference 3. The beginning-of-cycle (BOC) shutdown margin (SDM) is 1.32% delta-k, well in excess of the required 0.38% delta-k. The standby liquid control system (SLCS) was calculated to provide a SDM of 3.67% delta-k for cold conditions with all control rods in their full power positions. This fully meets shutdown requirements. Since these results have been obtained with previously approved methods and fall within the expected range, the staff concludes that the nuclear design of the Cycle 5 reload core is acceptable.

2.6 Transient Analyses

Core wide transients were analyzed with the COTRANSA computer code (Ref. 12) which includes a one-dimensional neutron kinetics model for evaluation of the axial power shape response during pressurization transients (generator

load rejection and feedwater controller failure). The referenced report has been reviewed by the staff and the methods for calculating the system transient response were found to be acceptable.

Calculation of the change in critical power ratio (CPR) during the core wide transient events involves the use of COTRANSA system results which serve as input to a XOBRA-T hot channel analysis model (Ref. 13) used to calculate the delta CPR values. The XCOBRA-T model has been reviewed by the staff and found to be acceptable. The licensee evaluated several categories of potential core wide transients for Cycle 5 and provided specific results for the three limiting transients: load rejection without bypass (LRWB), feedwater controller failure (FWCF), and loss of feedwater heating (LOFH). For operation at rated power in the range of EOC-2000 (3750)MWD/MTU to EOC, the LRWB is identified as the limiting transient. The calculated delta CPR, assuming WNP-2 measured scram speed, was 0.25 for ANF fuel and 0.29 for GE fuel resulting in MCPR limits of 1.31 and 1.35 for ANF fuel and GE fuel, respectively.

If the recirculation pump trip (RPT) should become inoperable, the limiting transient between 3750 MWD/MTU and EOC is still the LRWB. Assuming normal scram speeds, at end-of-cycle exposures, the MCPR operating limits are 1.38 (ANF fuel) and 1.42 (GE fuel) with an inoperable RPT. For technical specification scram times, the MCPR limits are 1.42 (ANF fuel) and 1.48 (GE fuel) with an inoperable RPT. These values are bounded by the proposed Cycle 5 MCPR operating limits and are, therefore, acceptable.

The control rod withdrawal error (CRWE) was found to be most limiting from BOC up to 3750 MWD/MTU. The delta CPR for the CRWE was 0.17 for both ANF fuel and GE fuel. At higher exposures, the CRWE delta CPR values are bounded by the LRWB transient as shown above.

The most limiting event for reactor vessel over-pressurization is the main steamline isolation valve (MSIV) closure without direct scram (single failure) on valve position. The maximum value of the sensed pressure in the steam dome was 1286 psig which corresponds to a maximum vessel pressure of 1315 psig at the lower plenum. These values are less than the Technical Specification limit of 1325 psig as measured by the steam dome pressure indicator and the 1375 psig ASME vessel pressure limit. This is acceptable.

The limiting plant system transients mentioned above were all analyzed at an increased core flow of 106% of rated core flow. ANF has performed analyses which demonstrate that the ANF 8x8C fuel assembly can operate satisfactorily from a mechanical standpoint at this increased flow. GE has also performed analyses for the reactor internals and for the GE fuel assembly which showed satisfactory operation at this increased flow. Based on these analyses and on the similarity between the two fuel types utilized in Cycle 5, the staff concludes that WNP-2 can operate safely with extended core flow up to 106% of rated core flow during Cycle 5.

The licensee reviewed the recirculation flow run-up analysis for Cycle 2 and concluded that the Cycle 2 analysis is applicable to Cycle 5 except for the six degree reduction in feedwater temperature at full power conditions. Thus, the reduced flow MCPR for Cycle 5 is changed on the conservative side from earlier cycles. The reduced flow MCPR operating limit for Cycle 5 presented in T/S Figure 3.2.3-1 is acceptable.

2.7 Postulated Accidents

The control rod drop accident (CRDA) yields a value of 121 cal/gm for the maximum deposited fuel rod enthalpy. This is well below the NRC required limit of 280 cal/gm and is, therefore, acceptable.

The loss of coolant accident (LOCA) analysis for Cycle 2 was performed for a full core of ANF 8x8C fuel and remains applicable for the Cycle 5 residual and reload ANF fuel. These LOCA analyses have covered an acceptable range of conditions, have been performed with approved methodology and the current technical specification MAPLHGR values for the ANF fuel were found acceptable. Since ANF 8x8C fuel is hydraulically and neutronically compatible with the GE fuel in Cycle 5, the existing GE LOCA analysis and MAPLHGR limits remain applicable to the GE fuel.

2.8 Single Loop Operation (SLO)

Single Loop Operation was approved in Amendment 62 for Cycle 4. The description and evaluation of SLO in Amendment 62 is applicable for Cycle 5 (Ref. 14, 15, 16) and thus, SLO is acceptable.

2.9 9x9-IX and 9x9-9X Lead Test Assemblies (LTAs)

The licensee intends to load the 9x9 lead fuel test assemblies (LTAs) in core locations which have been analyzed to have sufficient margin such that the LTAs are not expected to be the limiting assemblies in the core on neither a nodal or a bundle power basis. This approach is intended to prevent the 9x9 LTAs from ever being the limiting fuel bundles. Evaluations were performed consistent with ANF methodolgoy (Ref. 11) to establish a licensing basis for two ANF 9x9-IX and two ANF 9x9-9X LTA in the WNP-2 Cycle 5 core.

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

The dynamic response of the LTAs is expected to be almost identical to that of the 8x8 already in the core. This is due to the fact that the fuel assembly stiffness is provided by the assembly channel, which is the same in both designs. The mass of the LTAs is very close to that of the 8x8's. It thus follows that the dynamic response should be the same.

The 9x9 LTAs are hydraulically compatible with the co-resident ANF 8x8 fuel assemblies based on a comparison of fuel component hydraulic resistances. Steady state thermal hydraulic analysis has shown that even though the ANF 9x9 LTA design has a somewhat smaller flow area than the ANF 8x8 design, no reduction in thermal margin is expected in the Cycle 5 core. This is due to the increased critical power performance of the ANF 9x9 LTA design relative to the ANF 8x8 design at WNP-2 Cycle 5 conditions.

The average enrichment and enrichment distribution for the 9x9-IX and 9x9-9X fuel assemblies have been selected to match, as closely as possible, the neutronic performance of the four 8x8 XN-3 2.64 w/o U-235 reload assemblies included in the Cycle 5 reload. The fuel assembly average enrichment is 2.53 w/o U-235 for the 9x9-IX design and 2.59 w/o U-235 for the 9x9-9X design. The average enrichment of the 138 inch fuel assembly is 2.69 w/o U-235 for the 9x9-IX and 2.75 w/o U-235 for the 9x9-9X. Each 9x9 assembly contains six fuel rods containing Gd_2O_3 blended with 2.51 w/o U-235. The 9x9 fuel assembly contains 72 fuel rods and one central water channel displacing nine rod positions.

The nuclear characteristics of the 9x9 LTAs are similar to the characteristics of the ANF 8x8 fuel. The effect of replacing four ANF 8x8 assemblies with the four ANF 9x9 LFAs on the Cycle 5 core neutronics is negligible. The maximum cold uncontrolled non-voided k of the 9x9 fuel is 1.215 compared to the maximum k of 1.229 for the XN-3 8x8 fuel; thus the 9x9 fuel is compatible with the 8x8 fuel for fuel storage in the spent fuel pool.

Analyses of the WNP-2 Cycle 5 limiting transients have been performed for ANF 8x8, ANF 9x9 LTAs, and GE P8x8R fuel. It has been shown that using the XN-3 ANF CHF correlation, the bundle power required to produce transition boiling in an ANF 9x9 LTA is higher than that for an ANF 8x8 bundle. That is, when an ANF 9x9 LTA bundle is modeled as an 8x8 bundle with equivalent conditions, there is margin to the MCPR safety limit during all transients. The Cycle 5 Safety Limit Analysis considered the LTAs such that the MCPR safety limit of 1.06 is also applicable to the LTAs. Therefore, the ANF 9x9 LTAs can be monitored to the ANF 8x8 fuel limits.

Since heatup is primarily a planar and not an axial phenomena, the appropriate bundle power limit that is derived from a LOCA analysis is the peak bundle planar power. The ANF 9x9 LTAs have better cooling during LOCA conditions relative to an ANF 8x8 fuel assembly due to the lower stored energy in the fuel rods, a greater surface area provided by the larger number of fuel rods, and more inert surface from the central water channel. Thus, a LOCA analysis for the ANF 9x9 LTAs would yield lower Peak Cladding Temperatures (PCTs) and metal-water reactions than an ANF 8x8 assembly at the same bundle peak planar power. The MAPLHGR limits for the ANF 9x9 LTAs restrict the peak bundle planar power to that analyzed for the ANF 8x8 fuel and assure that the criteria are met for the ANF 9x9 LTAs in Cycle 5.

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

All operational limits used for ANF 8x8 fuel are applicable to the ANF 9x9 LTAs except for fuel type specific MAPLHGR limits and the 9x9-IX and 9x9-9X LHGR limits. The LHGR limits for the 9x9-IX and 9x9-9X LFAs are shown in T/S Figures 3.2.4-2 and 3.2.4-3 respectively, and the MAPLHGR limits for the LTAs are shown in T/S Figure 3.2.1-6. These are acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

(a) T/S 3.1.3.4: Average Scram insertion time for the four control rods are changed to agree with the values assumed in the analysis. The proposed change is acceptable.

(b) T/S 3/4.2.1: This specification is amended to include reference to Figure 3.2.1-6 giving the specific limits applicable to the lead fuel assemblies. The associated Bases section is similarly revised. As stated above under section 2.9, these changes are acceptable.

(c) T/S 3/4.2.3: Table 3.2.3-1 and Figure 3.2.3-1 have been revised to reflect Cycle 5 MCPR operating limits. These new limits are based on the Cycle 5 reload safety analysis which has been evaluated and approved in Section 2 and are, therefore, acceptable.

(d) T/S 3/4.2.4: This specification is amended to include reference to Figures 3.2.4-2 and 3.2.4-3, which give the limits for the linear heat generation rate applicable to the lead fuel assembkies. As stated above under section 2.9, these changes are acceptable.

(e) Bases Section 2.1.2: The third paragraph of this Bases section is revised to delete reference to Bases Table B2.1.2-2. Table B2.1.2-2 was removed by Amendment 28. Deletion of this reference is administrative and is acceptable.

(f) Bases section 3/4.2.1: This section is amended such that the reference cited in the final line of the section is identified. This is an editorial change and is acceptable.

(g) T/S 5.3: The description of the fuel assemblies in the reactor core is amended to show that the reload fuel may employ a nine by nine array of fuel rods. This change is to accommodate the lead fuel assemblies. Since the limits applicable to the lead fuel assemblies have been reviewed and found acceptable in Section 2.9, this change 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. 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 offsite, 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 STATE OFFICIAL

The Commission made a proposed determination that the amendment involves no significant hazards consideration (54 FR 13771, April 5, 1989) and consulted with the State of Washington. No public comments were received, and the State of Washington did not have any comment.

6.0 CONCLUSION

The staff has reviewed the reports submitted for the Cycle 5 reload of WNP-2 with ANF fuel using ANF methodology and analysis. Based on this review, the staff concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The 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. The proposed operation of WNP-2 for a fifth cycle including the use of four 9x9 LTAs, is therefore, acceptable. 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 (3) 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: George Thomas, SRXB

Dated: June 7, 1989

7.0 REFERENCES

1. Letters from G. C. Sorensen (WPPSS) to NRC, Request for Amendment to Technical Specifications - Reload License Amendment (Cycle 5), March 3, 1989, April 20, 1989 and June 1, 1989.

2. WPPSS-EANF-124, "WNP-2 Cycle 5 Reload Summary Report," February 1989.

3. ANF -89-02, "WNP-2 Cycle 5 Reload Analysis Report," Rev. 1, March 1989.

4. ANF-89-01, "WNP-2 Cycle 5 Plant Transient Analysis Report," Rev. 1, March 1989.

5. WPPSS-C-ANF-101, "WNP-2 Cycle 2 Reload Summary Report," February 1986.

6. XN-NF-81-21(A), Rev. 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, January 1982.

7. XN-NF-81-21(A), Revision 1, Supplement 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," March 1985.

8. XN-NF-85-67(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.

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

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

11. XN-NF-80-19(A), Volume 1 and Volume 1 Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," November 1981.

12. XN-NF-79-71(P), Revision 2 (as supplemented), "Exxon Nuclear Power Plant Transient Methodology," November 1981.

13. XN-NF-84-105(A), Volume 1 (as supplemented), "XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis," February 1987.

14. ANF-87-119, "WNP-2 Single Loop Operation Analysis," September 1987.

15. ANF-87-118, "WNP-2 LOCA Analysis for Single Loop Operation," September 1987.

16. WPPSS-EANF-115, "WNP-2 Single Loop Operation Summary Report," February 1988.