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. 59
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 7, 1988 and supplemented on April 12, 1988, 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 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. 59 , 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

Charles M. Trammell
for 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 9, 1988

ENCLOSURE TO LICENSE AMENDMENT NO.59

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.

REMOVE	INSERT
v	v
3/4 1-7	3/4 1-7
3/4 2-7	3/4 2-7
3/4 3-104	3/4 3-104
B 2-2	B 2-2
	B 2-4

The following corresponding overleaf pages are also provided:

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3/4 1-8

3/4 2-8

3/4 3-103

B 2-1

B 2-3

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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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	0.455
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.

Table 3.2.3-1

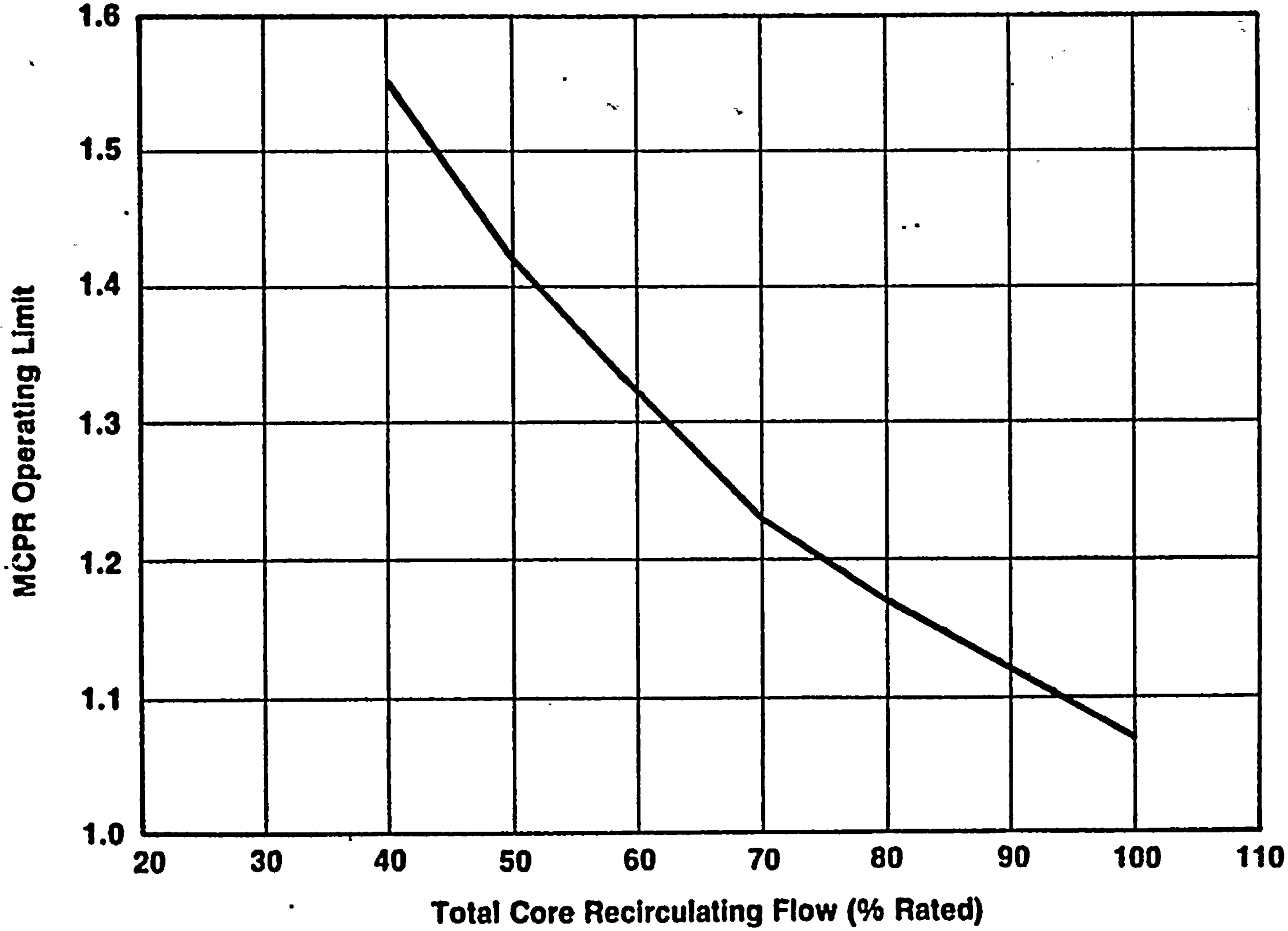
MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow	
			GE Fuel	ANF Fuel
1.	0 MWD/MTU - 3750 MWD/MTU	*	1.40	1.28
2.	3750 MWD/MTU - EOC MWD/MTU	Normal scram times**	1.40	1.31
3.	3750 MWD/MTU - EOC MWD/MTU	Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.50	1.38
4.	3750 MWD/MTU - EOC MWD/MTU	RPT inoperable Normal scram times	1.50	1.37
5.	3750 MWD/MTU - EOC MWD/MTU	RPT inoperable Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.55	1.43

*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.

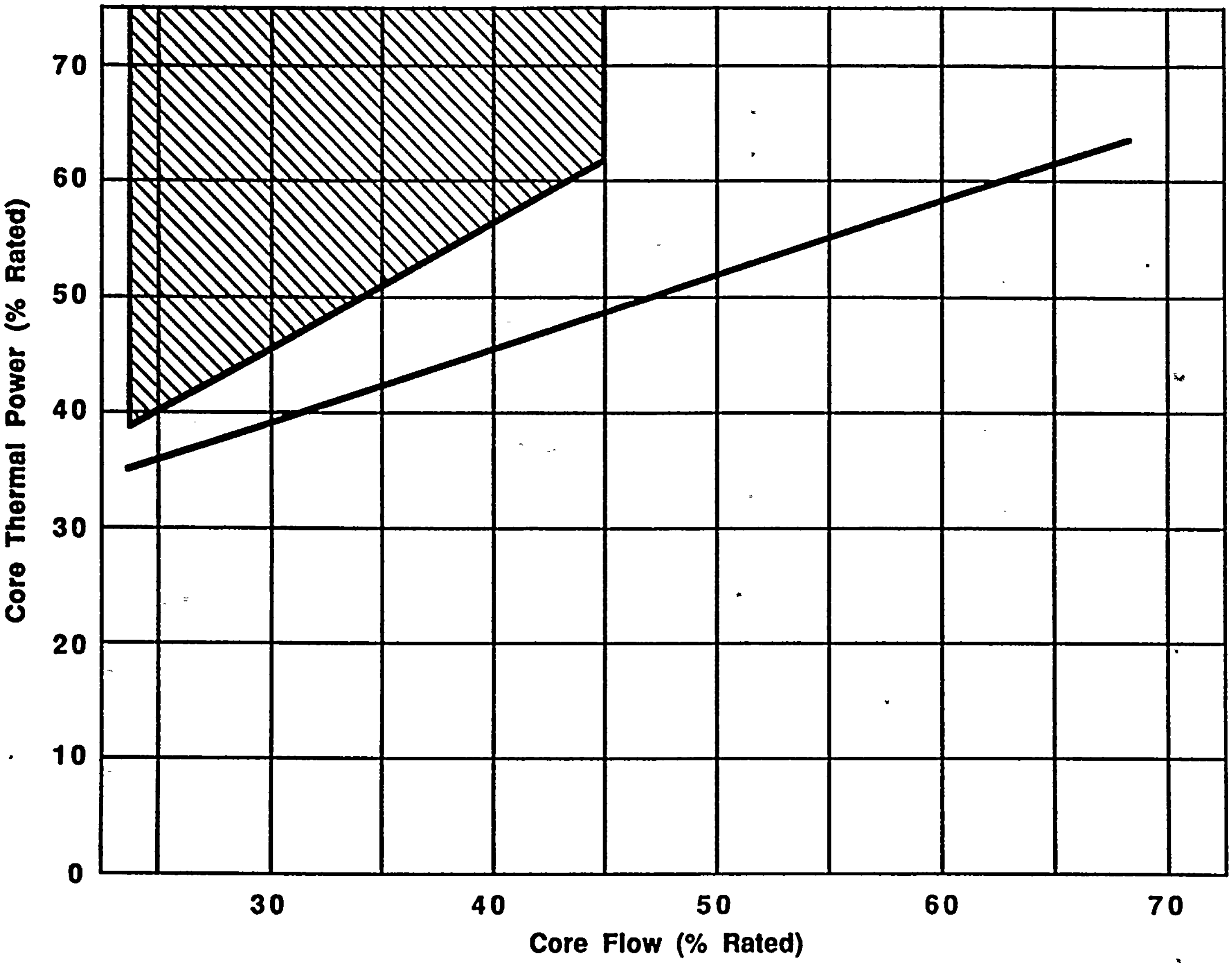
**These MCPR values are based on the ANF Reload Safety Analysis performed using the control rod insertion times shown below (defined as normal scram). In the event that surveillance 4.1.3.2 shows these scram insertion times have been exceeded, the plant thermal limits associated with normal scram times 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



Reduced Flow MCPR Operating Limit
Figure 3.2.3-1

860599.1A



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

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, 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-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.

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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.

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 $.25 \times 10^6$ 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 28×10^3 lbs/h (approximately a mass velocity of $.25 \times 10^6$ 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 28×10^3 lbs/h. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power

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BASES TABLE B2.1.2-1

UNCERTAINTIES CONSIDERED IN

THE MCPR SAFETY LIMIT

Parameter	STANDARD DEVIATION*
Feedwater Flow Rate	.0176
Feedwater Temperature	.0076
Core Pressure	.0050
Total Core Flow Rate	.0250
Core Inlet Enthalpy	.0024
XN-3 Critical Power Correlation	.0411
Assembly Flow Rate	.0280
Power Distribution:	
Radial Peaking Factor	.0528
Local Peaking Factor	.0246

* Fraction of Nominal Value.