June 9, 1988

Docket No. 50-397

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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. 59 TO FACILITY OPERATING LICENSE NO. NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 67541)

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 7, 1988 (G02-88-054) as supplemented on April 12, 1988 (G02-88-087).

This amendment revises Technical Specification Table 3.2.3-1, "MCPR Operating Limits," and Figure 3.3.10-1, "Thermal Power Limits of Specification 3.3.10-1." The revisions provide the operating limits established for the fourth fuel cycle of operation.

As part of this amendment we have also made corrections to Technical Specification pages v, 3/4 1-7, B 2-2, and B 2-4. These corrections were reviewed as part of previous fuel reload amendments but the page changes were inadvertently missed at the time.

A copy of the related safety evaluation supporting Amendment No. 59 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 and Special Projects

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PDR ADOCK 05000397
P PDR

Enclosures:
1. Amendment No. 59 to Facility Operating License No. NPF-21
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

DRSP/PDV	DRSP/PDV	*OGC	DRSP/D:PDV
JLee	RSamworth:cw	MYoung	GWKnighton
06/8/88	06/7/88	06/01/88	06/9/88

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

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cc w/enclosures:
See next page

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

WPPSS Nuclear Project No. 2
(WNP-2)

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Regional Administrator, Region V
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Prosser, Washington 99350

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.

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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 3-103

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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

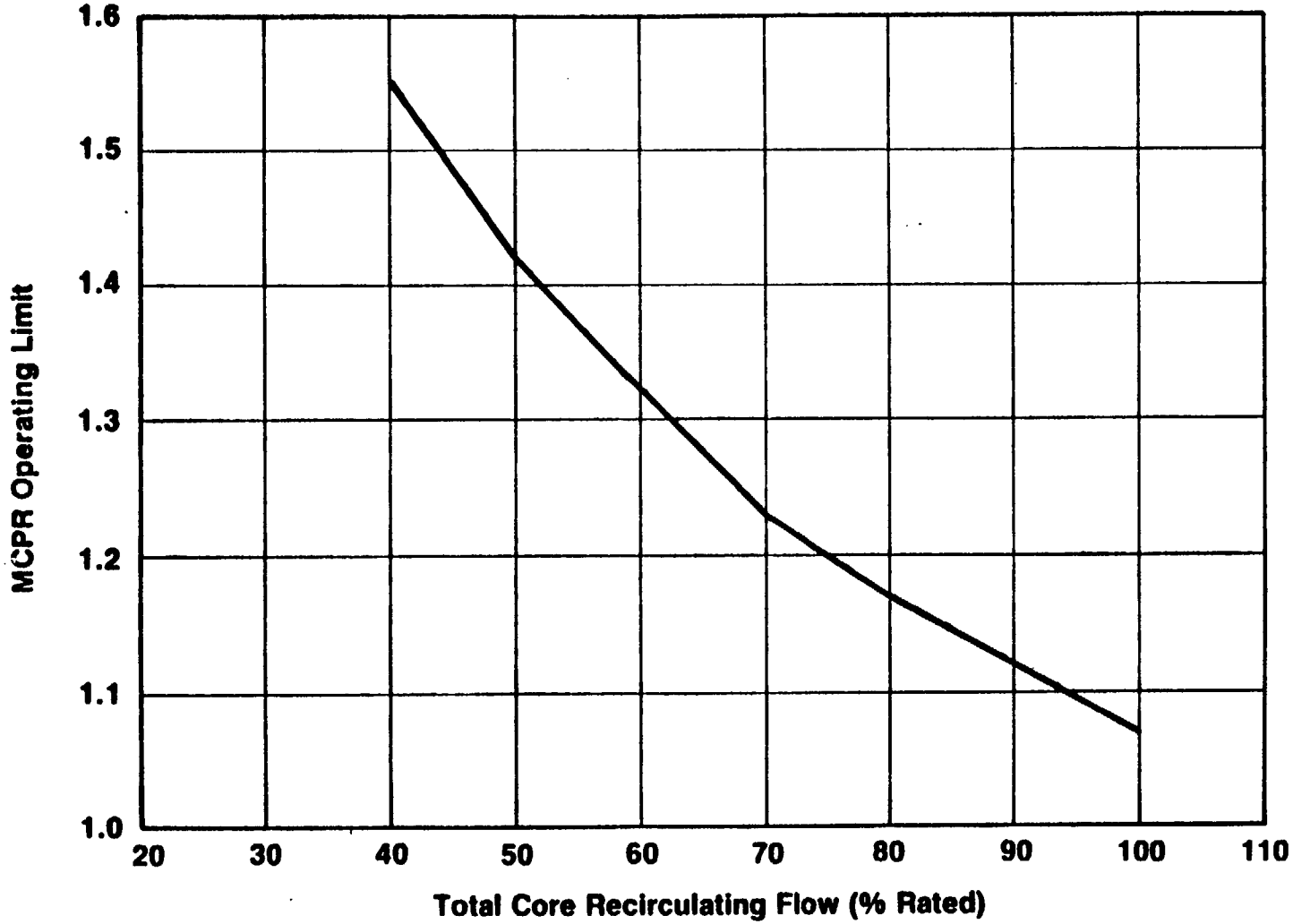
MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow	
			GE Fuel	ANF Fuel
1.	0 $\frac{MWD}{MTU}$ - 3750 $\frac{MWD}{MTU}$	*	1.40	1.28
2.	3750 $\frac{MWD}{MTU}$ - EOC $\frac{MWD}{MTU}$	Normal scram times**	1.40	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)	1.50	1.38
4.	3750 $\frac{MWD}{MTU}$ - EOC $\frac{MWD}{MTU}$	RPT inoperable Normal scram times	1.50	1.37
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)	1.55	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

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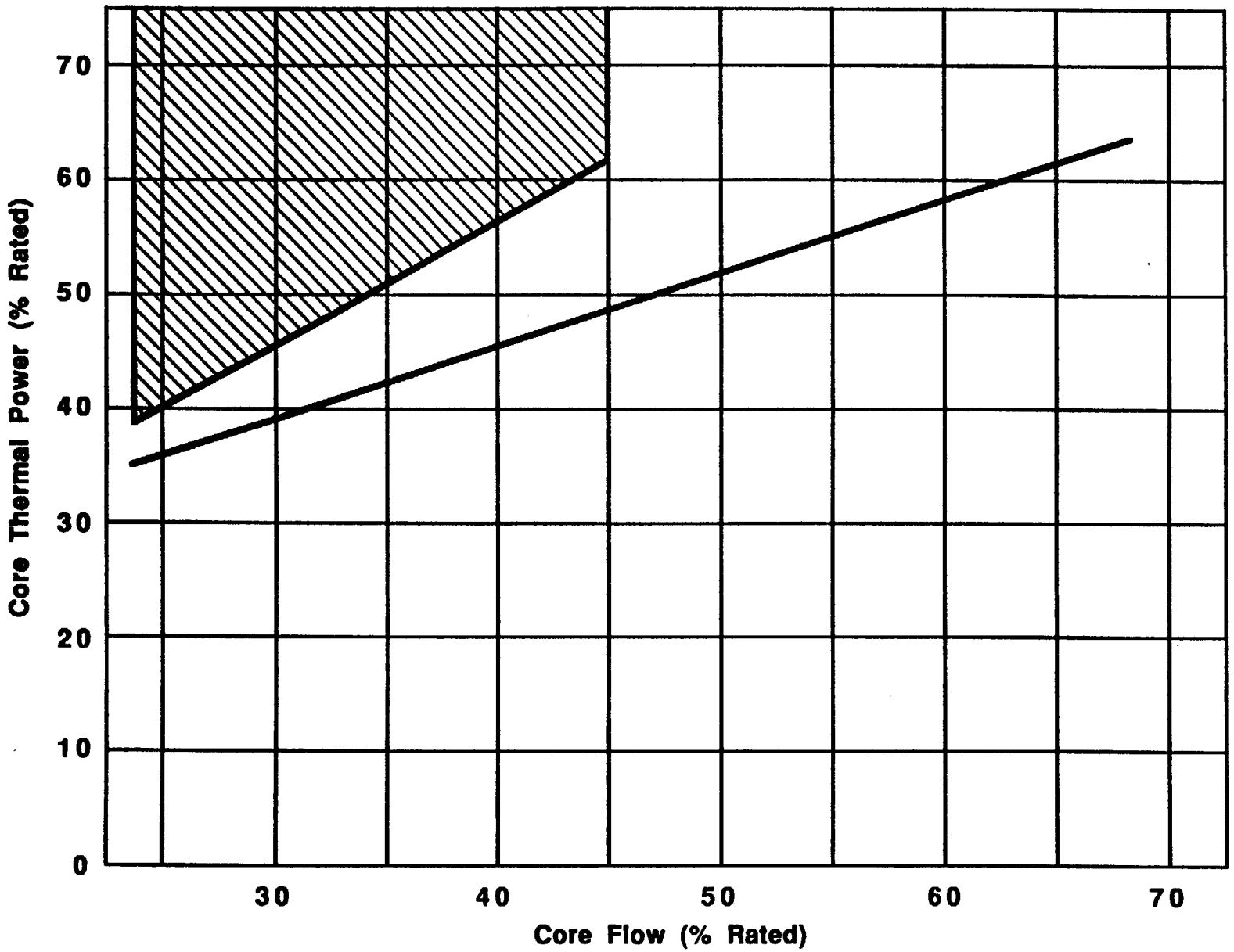

Thermal Power Limits of Specification 3.3.10-1
Figure 3.3.10-1

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.

---

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.59 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 dated March 7, 1988 (Ref. 1), Washington Public Power Supply System, the licensee, proposed to amend Facility Operating License NPF-21 to support Cycle 4 operation of their Nuclear Plant No. 2 (WNP-2) with Advanced Nuclear Fuels Corporation (ANF) reload fuel. In support of the Cycle 4 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 changes (Ref. 5). The proposed Technical Specification page changes applicable to this amendment were omitted from the March 7, 1988 letter but were provided by supplemental letter dated April 12, 1988. The proposed page changes included with the March 8 submittal were withdrawn by the April 12 submittal.

2.0 EVALUATION

2.1 Reload Description

The WNP-2 Cycle 4 reload will incorporate a total of 152 unirradiated ANF 8x8C fuel assemblies which replace 152 of the General Electric (GE) initial core fuel assemblies. Twenty-four of these assemblies have an average enrichment of 2.72 weight percent U-235 while the remaining 128 are enriched to an assembly average of 2.64 weight percent U-235. The remainder of the core is comprised of 148 ANF 8x8C assemblies loaded for Cycle 3, 128 ANF 8x8C assemblies loaded for Cycle 2 and 336 GE 8x8RP assemblies remaining from the initial core.

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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 4 core were approved for operation in previous cycles.

The 128 8x8C ANF reload fuel assemblies manufactured for loading in Cycle 4 are essentially identical to the 24 8x8C ANF reload assemblies originally fabricated for reload in Cycle 3 in all major physical characteristics except U-235 enrichment. Although minor differences in end plug design exist between these two assembly designs, they are essentially interchangeable. All of the reload fuel assemblies are essentially identical to the 8x8C ENC fuel approved for use in Cycle 2 (Ref. 9). Based on this, and on the fuel mechanical design analysis and results which used approved methodologies (Ref. 10), the staff finds the mechanical design of the ANF 8x8C reload fuel for the WNP-2 Cycle 4 reload acceptable.

2.3 Thermal-Hydraulic Design

The ANF thermal-hydraulic methodology and criteria used for the Cycle 4 design and analysis is the same as the previous WNP-2 reloads. These previous reviews 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 (MCPR) is acceptable. The methodology for Cycle 4 is based on ANF's revised critical power methodology (Ref. 11) 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. 12). Therefore, the proposed safety limit MCPR of 1.06 for all fuel types in this reload is acceptable.

The WNP-2 Technical Specifications include surveillance requirements for detecting and suppressing power oscillations. The staff has required that detect and suppress surveillance be used in regions which have code calculated decay ratios of 0.75 or greater and that operation be forbidden in regions having calculated decay ratios of 0.90 and greater. The NRC Generic Letter 86-02 addressed both GE and ANF stability calculation methodology and concluded that regions of potential instability constituted calculated decay ratios of 0.80 and greater using GE methodology and 0.75 and greater using the ANF methodology. Using the COTRAN code (Ref. 13), ANF has determined that the worst case value of decay ratio is less than 0.75 in the area of the power/flow map bounded by the APRM rod block line at 45% rated flow. In addition, the worst case decay ratio is no greater than 0.90 in the area of allowable low flow operation (detect and suppress region). The bounding power/flow points in the detect and suppress region are the APRM rod block line at 27.6% core flow (46%

power - minimum allowable two pump flow) and the APRM rod block line at 23.8% core flow (42% power - natural circulation flow). The COTRAN calculated decay ratios for these two state points were 0.88 and 0.82, respectively. The licensee's analysis included a 3% margin from the rod block line in order to bound future vendor stability calculations. The power boundary has been linearized between two points, (24% flow, 39% power) and (45% flow, 62% power). The staff finds the stability analyses and surveillance requirements acceptable.

2.4 Nuclear Design

The nuclear design for Cycle 4 has been performed with ANF methodologies previously reviewed and approved (Ref. 13). The fuel loading pattern is given in Figure 4.1 of Reference 3. The beginning-of-cycle (BOC) shutdown margin (SDM) is 1.06% 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 3.46% 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 4 reload core is acceptable.

2.5 Transient Analyses

Core wide transients were analyzed with the COTRANSA computer code (Ref. 14) 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 XCOBRA-T hot channel analysis model (Ref. 15) 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 4 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 was 0.24 for ANF fuel and 0.25 for GE fuel assuming normal scram speed resulting in MCPR limits of 1.30 and 1.31 for ANF fuel and GE fuel, respectively. With Technical Specification scram times, these values become 1.36 for ANF fuel and 1.38 for GE fuel. 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, the MCPR operating limits are 1.35 (ANF fuel) and 1.38 (GE fuel) with an inoperable RPT. For Technical Specification scram times, the MCPR limits are 1.41 (ANF fuel) and 1.44 (GE fuel) with an inoperable RPT. These values are bounded by the proposed Cycle 4 MCPR operating limits and are, therefore, acceptable.

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 licensee has also determined the limiting local transient to be the control rod withdrawal error (CRWE) and has calculated the MCPR operating limit as a function of the rod block monitor (RBM) setpoint. The CRWE was found to be most limiting from BOC up to 3750 MWD/MTU. The delta CPR for the CRWE with a 106% RBM setpoint was 0.17 (ANF fuel) and 0.21 (GE fuel), 0.18 (ANF fuel) and 0.22 (GE fuel) for a 107% RBM setpoint, and 0.20 (ANF fuel) and 0.23 (GE fuel) for a 108% RBM setpoint. Therefore, operation with a 108% RBM setting requires a MCPR limit of 1.26 for ANF fuel and 1.29 for the GE fuel which are bounded by the proposed Cycle 4 MCPR operating limits between BOC and 3750 MWD/MTU. At higher exposures, the CRWE delta CPR values are bounded by the LRWB transient as shown above.

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 (Ref. 16). GE has also performed analyses for the reactor internals and for the GE fuel assembly which showed satisfactory operation at this increased flow (Ref. 17). Based on these analyses and on the similarity between the two fuel types utilized in Cycle 4, the staff concludes that WNP-2 can operate safely with extended core flow up to 106% of rated core flow during Cycle 4. Thus, this increased core flow is acceptable.

2.6 Postulated Accidents

The control rod drop accident (CRDA) yields a value of 149 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 4 residual and reload ANF fuel. These LOCA analyses have covered an acceptable range of conditions, have been performed with approved

methodology and the resulting Technical Specification MAPLHGR values for the ANF fuel are acceptable. Since ANF 8x8C fuel is hydraulically and neutronically compatible with the GE fuel in Cycle 4, the existing GE LOCA analysis and MAPLHGR limits remain applicable to the GE fuel.

3.0 TECHNICAL SPECIFICATION CHANGES

Specification 3/4.2.3: Table 3.2.3-1 and Figure 3.2.3-1 have been revised to reflect Cycle 4 MCPR operating limits. These new limits are based on the Cycle 4 reload safety analysis which has been evaluated and approved in Section 2 and are, therefore, acceptable.

The proposed changes on Table 3.2.3-1 included new operating limits which would apply if the staff were to approve separate license amendment applications for feedwater temperature reduction and for extended conditions for single loop operations. Since this Safety Evaluation Report does not address those separate amendment applications, the new operating limits are not included at this time.

4.0 CORRECTIONS TO THE TECHNICAL SPECIFICATIONS

The license amendment application for the cycle two refueling included several proposed changes which were reviewed by the staff and approved (Amendment 28 issued May 23, 1986) but were omitted from the license amendment as issued.

The licensee applied for removal of section 3/4.1.3.3, Control Rod Average Scram Insertion Times, by letter dated April 24, 1986 (G02-86-367). This deletion also necessitated revision of page v, of the table of contents. The Safety Evaluation issued with Amendment 28 found removal of this specification acceptable, However, the revised pages were not included in the Amendment when issued. Therefore this amendment deletes Section 3/4.1.3.3 and revises page v.

At three places on Table 3.2.3-1 (pages 3/4 2-7) of the Technical Specifications, specification 3.1.3.4 is referred to and the corresponding page number is indicated as page 3/4 1-7. The correct page number is 3/4 1-8. This correction is also made by this amendment.

The licensee requested deletion of Table B 2.1.2-2 (Page B 2-4) by letter dated February 26, 1988 (G02-86-173). Amendment 28 instructed the licensee to remove Page B 2-4. However no replacement page was issued. To provide continuity in the page numbers, a blank page B 2-4 should be inserted in the Technical Specifications.

Page B 2-2 is revised to correct two typographical errors introduced with the issuance of Amendment Number 45. On the fifth line from the bottom of the page and in footnote b, the correct number of the report providing the basis for the uncertainity in the XN-3 correlation is XN-NF-512(A), Rev. 1.

5.0 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 tha 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.

6.0 CONTACT WITH STATE OFFICIAL

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (53 FR 15920) on May 4, 1988 and consulted with the State of Washington. No public comments were received and the State of Washington did not have any comment.

7.0 CONCLUSION

The staff has reviewed the reports submitted for the Cycle 4 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 fourth cycle 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: Laurence I. Kopp

Dated: June 9, 1988

8.0 REFERENCES

1. Letter from G. C. Sorensen (WPPSS) to NRC, Request for Amendment to Technical Specifications - Reload License Amendment (Cycle 4), March 7, 1988.

2. WPPSS-EANF-119, "WNP-2 Cycle 4 Reload Summary Report," February 1988.

3. XN-NF-88-02, "WNP-2 Cycle 4 Reload Analysis Report," January 1988.

4. XN-NF-88-01, "WNP-2 Cycle 4 Plant Transient Analysis Report," January 1988.

5. Attachment to Reference 1, Proposed Changes to WNP-2 Technical Specifications.

6. ANF-87-129(P), "WNP-2 Reload XN-3(WPB3), Cycle 4 Design Report," December 1987.

7. XN-NF-81-21(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," September 1982.

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

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

10. XN-NF-85-67(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," July 1985.

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

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

13. 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.

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

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

16. Letter from J. B. Edgar (ENC) to WPPSS, Supplemental Analysis Results, ENNP-86-0067, April 15, 1986.

17. NEDC-31107, "Safety Review of WPPSS Nuclear Project No. 2 at Core Flow Conditions Above Rated Flow Throughout Cycle 1 and Final Feedwater Temperature Reduction," February 1986.