June 3, 1991

Docket No. 50-397

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

Dear Mr. Sorensen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. NPF-21 FOR THE WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 79884)

The Commission has issued the enclosed Amendment No. 92 to the Facility Operating License for the WPPSS Nuclear Project No. 2. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated February 28, 1991, as supplemented by letters dated March 21, and April 26, 1991.

The amendment modifies the facility minimum critical power ratio safety limit and associated bases to reflect cycle specific safety analyses resulting from use of a new reload methodology and effects of channel box bow phenomena.

A copy of the related Safety Evaluation is also enclosed. A notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:

Patricia L. Eng, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.92 to NPF-21
2. Safety Evaluation

cc w/enclosures:
See next page

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

WPPSS Nuclear Project No. 2
(WNP-2)

cc:
Mr. J. W. Baker
WNP-2 Plant Manager
Washington Public Power Supply System
P.O. Box 968, MD 927M
Richland, Washington 99352

G. E. C. Doupe, Esq.
Washington Public Power Supply System
3000 George Washington Way
P. O. Box 968
Richland, Washington 99532

Mr. R. G. Waldo, Chairman
Energy Facility Site Evaluation Council
Mail Stop PY-11
Olympia, Washington 98504

Mr. Alan G. Hosler, Licensing Manager
Washington Public Power Supply System
P. O. Box 968, MD 956B
Richland, Washington 99352

Mr. A. Lee Oxsen, Acting
Managing Director for Operations
Washington Public Power Supply System
P. O. Box 968, MD 1023
Richland, Washington 99352

Mr. Gary D. Bouchey, Director
Licensing and Assurance
Washington Public Power Supply System
P. O. Box 968, MD 280
Richland, Washington 99352

Regional Administrator, Region V
U.S. Nuclear Regulatory Commission
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

Chairman
Benton County Board of Commissioners
P. O. Box 190
Prosser, Washington 99350-0190

Mr. R. C. Sorensen
U. S. Nuclear Regulatory Commission
P. O. Box 69
Richland, Washington 99352

Nicholas S. Reynolds, Esq.
Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

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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. 92
License No. NPF-21

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

   A. The application for amendment by the Washington Public Power Supply System (licensees) dated February 28, 1991, and supplemented by letters dated March 21, 1991 and April 26, 1991, 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 rules and 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;

   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. 92 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

James E. Dyer

James E. Dyer, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 3, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 92

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. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
2-1	2-1
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC with two recirculation loop operation and shall not be less than 1.08 up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC with two recirculation loop operation or less than 1.08 up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC with single recirculation loop operation and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

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 1.07 up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC for two recirculation loop operation and 1.08 up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC for single recirculation loop operation for all nuclear fuel in WNP-2. MCPR greater than 1.07 up to 4500 MWD/MTU cycle exposure and 1.11 for cycle exposure greater than 4500 MWD/MTU to EOC for two recirculation loop operation and 1.08 up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC 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: ANF-524(P)(A), Rev. 2; ABB Atom Report UK90-126; GE11 Lead Fuel Assembly Report for Washington Public Power Supply System Nuclear Project No. 2, Reload 5, Cycle 6). The latter two references support application of the above established safety limit to GE11 and SVEA-96 LFA fuel in WNP-2.

2.1 SAFETY LIMITS

2.1.1. THERMAL POWER, Low Pressure or Low Flow

For certain conditions of pressure and flow, the ANFB correlation is not valid for all critical power calculations. The ANFB correlation is not valid for bundle mass velocities less than 0.10 x 10^6 lbs/hr-ft^2 or pressures less than 590 psia. 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

THERMAL POWER, Low Pressure or Low Flow (Continued)

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 0.25 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 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 785 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 ANFB correlation. The ANFB 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 reactor system and fuel uncertainties are given in ANF-524(P)(A), Rev. 2[a]. 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. Advanced Nuclear Fuels Critical Power Methodology for Boiling Water Reactors, ANF-524(P)(A), Rev. 2.

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
Assembly Flow Rate	.0280
Power Distribution:	
Radial Assembly Power	.0409
Local Power**	.0229
ANFB Correlation Additive Constants	.0200

*Fraction of Nominal Value.
**Relative Local Rod Power.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated February 28, 1991 (Ref. 1), as amended by letters dated March 21, 1991 (Ref. ?) and April 26, 1991 (Ref. 5), Washington Public Power Supply System (WPPSS), the licensee for WNP-2, proposed changes to the Technical Specification (TS) associated with the minimum critical power ratio (MCPR) Safety Limit for WNP-2. The licensee had concluded from their reload analysis that the reuse of channel boxes in the upcoming cycle, cycle 7, would require recalculation of the MCPR Safety Limit due to channel box bow effects. All calculations were carried out by an NRC approved methodology, (Ref. 3).

The proposed changes would modify the MCPR Safety Limit TS from 1.04 to 1.07 up to a cycle exposure of 4500 MWD/MTU and 1.11 from 4500 MWD/MTU to end of cycle (EOC) for two recirculation loop operation, and 1.08 up to 4500 MWD/MTU cycle exposure and 1.12 for cycle exposure greater than 4500 MWD/MTU to EOC with single recirculation loop operation.

This safety evaluation covers the staff review of the Washington Public Power Supply System amendments to TS 2.1.2 and Bases 2.0, reflecting changes to the MCPR safety limit in the upcoming cycle 7.

The March 21 and April 26, 1991, letters provided clarifying information which did not change the scope of the amendment request and did not affect the staff's initial determination of no significant hazards consideration.

2.0 EVALUATION

The calculation of the Safety Limit MCPR (SLMCPR) is based on statistical consideration of measurement and associated uncertainties with the thermal hydraulic state of the reactor using design basis radial, axial, and local power distribution and considering fuel assembly channel box bow, (Ref. 3). In calculating the SLMCPR, the licensee included the effects of fuel dependent parameters associated with a mixed core. Similarly, when a reload batch (from a specific vendor) is used to replace a group of dissimilar fuel assemblies, the core average fuel dependent parameters change because of the difference in the relative number of each type of the bundle in the core. This was accounted for in the SLMCPR calculation.

The licensee also utilized data pertaining to radial, axial and local peaking factors, from previous cycle in their evaluation of the SLMCPR. Available operating data for WNP-2 and the predicted operating conditions for cycle 7 were evaluated to identify the design basis power distributions for use in the cycle 7 MCPR Safety Limit calculation.

The licensee conducted some 500 Monte Carlo trials to determine that for a minimum CPR value of 1.07 at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition with a confidence level of 95% for the design basis power distributions from BOC to a cycle average burnup of 4500 MWD/MTU. Similarly, to provide the same protection, a minimum CPR value of 1.11 is required for the design basis power distributions from 4500 MWD/MTU to EOC. For single loop operation, the single loop SLMCPR is increased by .01 to account for additional total core flow uncertainties to 1.08 and 1.12 for below and above 4500 MWD/MTU, respectively.

The above calculations included the effects of channel box bow on the Safety Limit MCPR. Without channel box bow effects, the SLMCPR would have been reduced by about 0.03. The Supply System will reuse some initial core channel boxes on ANF 8x8 fuel assemblies in the WNP-2 cycle 7 core. The effects of reused channel boxes adjacent to assemblies with exposed channel boxes was also included as input data in the 8x8 fuel calculations. The input data for the 9x9-9X fuel type, was based on a maximum channel box bow assuming a new channel on the 9x9-9X fuel adjacent to assemblies with exposed channels.

The staff has reviewed the licensee's submittal and has found that the proposed TS changes to the Safety Limit MCPR values for Cycle 7 reload are acceptable since the approved methodology was used and the results are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Washington State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the 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 the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: T. Huang

Date: June 3, 1991

6.0 REFERENCES

1. G02-91-040, February 28, 1991, G.C. Sorensen, WPPSS, to USNRC, "Nuclear Plant No. 2, Operating License NPF-21, Request for Amendment to Technical Specifications Safety Limit; Thermal Power, High Pressure and High Flow."

2. G02-91-054, March 21, 1991, G.C. Sorensen, WPPSS to USNRC, "Nuclear Plant No. 2, Operating License NPF-21, Request for Amendment to Technical Specifications Safety Limit; Thermal Power, High Pressure and High Flow - Revision."

3. Letter, August 8, 1990, A.C. Thadani, USNRC to R.A. Copeland, ANF, "Acceptance for referencing of Topical Report ANF-524(P), Revision 2, ANF Critical Power Methodology for Boiling Water Reactor."

4. Letter, April 19, 1991, P.L. Eng, USNRC to G.C. Sorensen, WPPSS, "Request for Additional Information Regarding Request for Amendment to the Technical Specification Safety Limit: Thermal Power, High Pressure and High Flow."

5. Letter, April 26, 1991, G.C. Sorensen, WPPSS to USNRC, "Nuclear Plant No. 2, Operating License NPF-21 Request for Amendment to Technical Specifications Safety Limit; Thermal Power and High Flow - Additional Information (TAC No. 79884)."