March 1, 1990

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. 77 TO FACILITY OPERATING LICENSE NO. NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 66885)

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 December 15, 1987, as supplemented and amended by letters dated March 7, 1988, April 12, 1988, September 14, 1988, March 3, 1989, April 20, 1989, June 1, 1989, and February 14, 1990, which presented supplementary and clarifying information to the NRC but did not change the nature of the amendment request.

This amendment adds a new section 3/4.1.6, "Reactivity Control Systems, Feedwater Temperature" which specifies that feedwater temperature shall not be reduced below 355°F for the purpose of fuel cycle extension. The amendment revises the MCPR Operating Limits in Table 3.2.3-1 by adding limits which would apply at the end of the fuel cycle when feedwater temperature is to be reduced. The amendment also adds definitions and revises the bases to cover feedwater temperature reduction.

A copy of the related safety evaluation supporting the amendment is enclosed. A Notice of Issuance will be published in the Federal Register .

Sincerely,

original signed by Robert 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. 77 to Facility Operating License No. NPF-21
2. Safety Evaluation

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

| | | | |
|---|---|---|---|
| DRSP/PD5 | DRSP/PD5* | OGC* | DRSP/AD:PD5 |
| PShea | RSamworth:dr | MYoung | CMTrammell |
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[WNP2 AMD 66885]

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

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2. Safety Evaluation

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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. 77
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 December 15, 1987, as supplemented by letters dated March 7, 1988, April 12, 1989, September 14, 1988, March 3, 1989, April 20, 1989 June 1, 1989, and February 14, 1990, 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) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 77 , 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 III

Charles M. Trammell III, Acting Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 1, 1990

ENCLOSURE TO LICENSE AMENDMENT NO. 77

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.

| AMENDMENT PAGE | OVERLEAF PAGE |
|---|---|
| i | -- |
| ii* | -- |
| iii* | iv |
| v | vi |
| xii | xi |
| 1-3 | 1-4 |
| 3/4 1-23 | -- |
| 3/4 2-7 | -- |
| 3/4 2-8 | -- |
| B3/4 1-5 | -- |
| B3/4 2-4 | B3/4 2-3 |
| B3/4 2-5* | -- |
| B3/4 2-6* | -- |

*Text was shifted on these pages but there was no change to the content of the text.

# INDEX

## DEFINITIONS

SECTION

# INDEX

## DEFINITIONS

SECTION

SECTION

DEFINITIONS (Continued) PAGE

# INDEX

## SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS



## BASES

## INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS



I

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

END OF CYCLE (EOC)

1.12A The END-OF-CYCLE (EOC) shall be the core exposure at which rated thermal power, rated core flow, and rated feedwater temperature would all be achieved if all control rods were fully withdrawn.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to energization of the recirculation pump circuit breaker trip coil from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:

a. Turbine throttle valves channel sensor contact opening, and
b. Turbine governor valves initiation of valve fast closure.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FINAL FEEDWATER TEMPERATURE REDUCTION (FFTR)

1.13A FINAL FEEDWATER TEMPERATURE REDUCTION (FFTR) shall be operation at or beyond EOC for the purpose of extending the normal fuel cycle by plant operation with a final feedwater temperature reduced from the normal rated power temperature condition.

FRACTION OF LIMITING POWER DENSITY

1.14 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

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LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.21 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.22 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.23 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be highest value of the FLPD which exists in the core.

MAXIMUM TOTAL PEAKING FACTOR

1.24 The MAXIMUM TOTAL PEAKING FACTOR (MTPF) shall be the largest TPF which exists in the core for a given class of fuel for a given operating condition.

MEMBER(S) OF THE PUBLIC

1.25 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.26 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and in the conduct of the environmental radiological monitoring program.

## 3/4.1.6 FEEDWATER TEMPERATURE

### LIMITING CONDITION FOR OPERATION

3.1.6 For the purposes of cycle extension, the feedwater temperature entering the reactor vessel shall not be reduced to less than 355°F.

APPLICABILITY: OPERATIONAL CONDITION 1, after the EOC exposure has been achieved with steady state THERMAL POWER greater than or equal to 47% of RATED THERMAL POWER.

ACTION:

With feedwater temperature entering the reactor vessel at a value below 355°F, initiate corrective action within 15 minutes and restore feedwater temperature 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

4.1.6 During cycle operation beyond EOC exposure, the feedwater temperature entering the reactor vessel shall be determined to be greater than or equal to 355°F:

a. At least once per 24 hours, and

b. Initially after establishing a reduced feedwater temperature lineup.

Table 3.2.3-1
MCPR OPERATING LIMITS

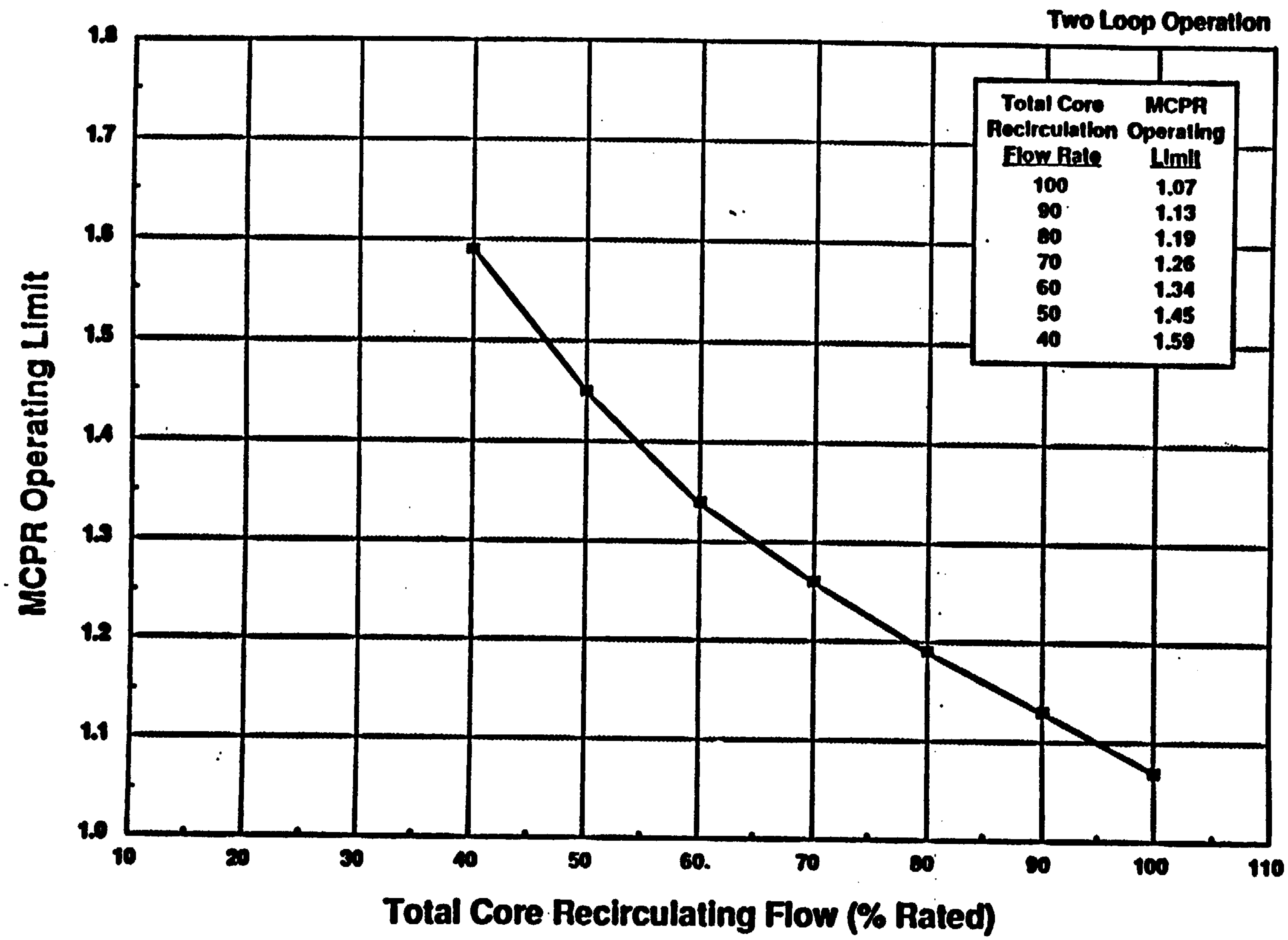
| | Cycle Exposure | Equipment Status | MCPR Operating Limit Up to 106% Core Flow | |
|---|---|---|---|---|
| | | | GE Fuel | ANF Fuel |
| 1. | 0 MWD/MTU - 3750 MWD/MTU | * | 1.24 | 1.24 |
| 2. | 3750 MWD/MTU - EOC MWD/MTU*** | Normal scram times** | 1.35 | 1.31 |
| 3. | 3750 MWD/MTU - EOC MWD/MTU***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

***For Final Feedwater Temperature Reduction rated conditions beyond all rods out point, add 0.02 to the MCPR for both GE and ANF fuel.

| 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

**Reduced Flow MCPR Operating Limit**
**Figure 3.2.3-1**

## 3/4.1.6 FEEDWATER TEMPERATURE

For the purpose of extending the cycle, feedwater temperature may be used for reactivity addition to compensate for the reactivity loss due to fuel depletion. The analysis performed is applicable to core flow values up to the maximum attainable (106 percent of rated core flow) and to feedwater temperature reductions to as low as 355°F. It is anticipated that a thermal coastdown from rated power with feedwater temperature held at 355°F would follow the rated run. This analysis also supports thermal coastdown followed by feedwater temperature reduction if this order is desirable. The analysis covers a reduction in power by thermal coastdown to 47 percent of rated thermal power with feedwater temperature held at or above 355°F.

It should be noted that during a normal feedwater lineup, a feedwater temperature at 355°F entering the reactor vessel is achieved at approximately 47 percent of rated thermal power. The Limiting Condition for Operations clearly does not apply during reactor startups and shutdowns when reactor power is below the point at which a feedwater temperature of 355°F is attainable with a normal feedwater system lineup.

Prior to reaching the end-of-cycle exposure, operation with an abnormal feedwater heater lineup is permissible as the short-term effect of increased subcooling is to more strongly bottom peak the axial power shape allowing a scram to suppress the flux faster. Compensation for the long-term effect of a pronounced bottom burn can be made by rod pattern adjustments and axial flux shape monitoring.

BASES

## 3/4.2.3 MINIMUM CRITICAL POWER RATIO

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

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

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

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

MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

At EOC during FFTR, the LOAD REJECTION WITHOUT BYPASS transient is slightly more severe when compared to the same transient without FFTR, which is accounted for by an increased MCPR operating limit. The analysis conservatively reduces the feedwater temperature by 65°F and burns the produced power shape to achieve the final core conditions used in the transient analysis. This depletion causes the power peak to shift upwards, slightly increasing the time required for the normal scram to suppress the flux.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

3/4.2.6 POWER/FLOW INSTABILITY

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow).

In February, 1984, GE issued SIL 380 addressing boiling instability and supplying several recommendations. In this SIL, the power/flow map was divided into several regions of varying concern. It also discussed the objectives and philosophy of "detect and suppress," coining the phrase.

The ANF topical report for COTRAN (XN-NF-691P) discusses boiling instability. The SER written on this topical (dated May 10, 1984) interprets the topical to require that the detect-and-suppress surveillance be used in regions which have code calculated decay ratios .75 or greater and that operation is forbidden in regions having calculated decay ratios of .9 and greater.

POWER/FLOW INSTABILITY (Continued)

The NRC Generic Letter 86-02 addressed both GE and ANF (then EXXON) stability calculation methodology and stated that due to uncertainties, General Design Criterias 10 and 12 could not be met using analytic procedures on a BWR 5 design. The letter espoused GE SIL 380 and stated that General Design Criterias 10 and 12 could be met by imposing the SIL 380 recommendations in operating regions of potential instability. The NRC concluded that regions of potential instability constituted calculated decay ratios of .8 and greater by the GE methodology and .75 and greater by the EXXON methodology.

Predicated on the SIL 380 endorsement, WNP-2 has divided the power/flow map on the following boundary lines:

1. 80% rod line
2 45% core flow line
3. 100% rod line
4. Natural Circulation flow line
5. Minimum Forced Circulation for normal recirculation lineup.

This division conforms to the SIL 380 recommendations. For LCO 3.2.6, the region of concern (Region A) is bounded by the more conservative of either the 100% rodline or a line defining a calculated decay ratio of 0.9, the natural circulation flow line, and the 45% core flow line. Calculated decay ratios outside Region A must be less than 0.9. Operation in the region between the two flow lines and above the more conservative of either the 100% rodline or a line defining a calculated decay ratio of 0.9 is forbidden due to the potential for boiling instabilities.

3/4.2.7 STABILITY MONITORING - TWO LOOP OPERATION

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod patterns, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux signal decay ratios should be monitored while operating in this region.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.75 was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a thermal power greater than that corresponding to the 80% rodline.

Stability monitoring is performed utilizing the ANNA system. The system shall be used to monitor APRM and LPRM signal decay ratio and peak-to-peak

STABILITY MONITORING - TWO LOOP OPERATION (Continued)

noise values when operating in the region of concern. A minimum number of LPRM and APRM signals are required to be monitored in order to assure that both global (in-phase) and regional (out-of-phase) oscillations are detectable. Decay ratios are calculated from 30 seconds worth of data at a sample rate of 10 samples/second. This sample interval results in some inaccuracy in the decay ratio calculation, but provides rapid update in decay ratio data. A decay ratio of 0.75 is selected as a decay ratio limit for operator response such that sufficient margin to an instability occurrence is maintained. When information shall be continuously calculated and displayed. A surveillance requirement to continuously monitor decay ratio and peak-to-peak noise values ensures rapid response such that changes in core conditions do not result in approaching a point of instability.

3/4.2.8 STABILITY MONITORING - SINGLE LOOP OPERATION

The basis for stability monitoring during single loop operation is consistent with that given above for two loop operation. The smaller size of the region of allowable operation, Region C, is due to a limit on the allowed flow above the 80% rodline. When operating above the 80% rodline in single loop operation, the core flow is required to be greater than 39%.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.77 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 (Ref. 1) dated December 15, 1987, Washington Public Power Supply System (the licensee) proposed an amendment to the Technical Specifications to allow operation of Washington Nuclear Project, Unit 2 (WNP-2) with final feedwater temperature reduction (FFTR) and subsequent thermal coastdown to 65% power in order to increase electrical generation by extending the operating cycle beyond a 12-month fuel cycle. The changes to the Technical Specifications include: (1) addition of a section which limits feedwater temperature when temperature is being reduced to extend the fuel cycle, (2) change to the minimum critical power ratio (MCPR) limits in Table 3.2.3-1, to specify the operating limits avoiding plant operation in the region which may result in a fuel failure during transients, and (3) changes to bases and definitions in the Technical Specifications to incorporate updated descriptions. By letters dated September 14, 1988 (Ref. 7) and February 14, (Ref. 13) 1990 the request was supplemented with supporting information requested by the staff. By letters dated March 7, 1988 (ref. 9) and April 12, 1988 (ref. 10), the licensee submitted an application for the Cycle 4 fuel reload. That reload application included limits for FFTR at the end of Cycle 4.

By letter dated March 3, 1989, (Ref. 8) April 20, 1989 (Ref. 11) and June 1, 1989 (Ref. 12) the licensee submitted analyses supporting a request to amend the license for the fifth cycle of operation. The support material included analyses of safety limits for final feedwater temperature reduction.

In support of these proposed changes, the licensee submitted a General Electric (GE) report, NEDC-31107, entitled "Safety Review of WPPSS Nuclear Project No. 2 at Core Flow Conditions Above Rated Flow Throughout Cycle 1 and Final Feedwater Temperature Reduction." The GE report presents the results of a safety and impact evaluation for the operation of the WNP-2 with FFWTR and increased core flow (ICF). The limiting normal operational transients described in the Final Safety Analysis Report (FSAR) have been reevaluated. The loss-of-coolant accident, fuel loading error accident, rod drop accident, and rod withdrawal error event were also re-evaluated. In addition, the effect of increased pressure differences on the reactor internals components, fuel channels and fuel bundles was also analyzed to show that the design limits will not be exceeded. The effect of ICF on the flow-induced vibration response of the reactor internals was also evaluated to ensure that the response is within acceptable limits. The thermal-hydraulic stability was evaluated for ICF/FFWTR operation, and the increase in the feedwater nozzle and feedwater sparger usage factors due to the feedwater temperature reduction was determined. The impact of ICF/FFWTR operation on the containment LOCA response was also analyzed. The staff has reviewed the proposed changes to the Technical Specifications and the supporting analytical results (Refs. 2 through 4 and 7) and has prepared the following evaluation.

## 2.0 EVALUATION

WNP-2 is currently operating in Cycle 5 with a mixed core consisting of GE and Advanced Nuclear Fuels Corporation (ANF-formerly Exxon Nuclear) fuels. Since the ANF fuel has operated fewer cycles than the GE fuel, the ANF fuel assemblies have a higher power peaking factor due to lower burndown effect. However, the licensee performed safety analyses based on both types of fuels provided by ANF and GE to establish the operating MCPR limits for each type of the fuel to support the changes to the Technical Specifications submitted for NRC review and approval. The objective of the staff review is to confirm that the thermal-hydraulic design of the core has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during transient conditions, and is not susceptible to thermal-hydraulic instability.

The staff has reviewed the following areas: (1) the safety MCPR limit, (2) the operating MCPR limits (3) thermal-hydraulic stability, (4) reactor vessel beltline materials, (5) thermal fatigue stresses to feedwater nozzles, (6) reactor internals load impact, (7) flow-induced vibration, (8) feedwater nozzle and feedwater sparger fatigue usage, and (9) the proposed changes to the Technical Specification.

### 2.1 Safety MCPR Limit

The safety MCPR limit has been imposed to assure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during transients. A safety MCPR limit of 1.06 was previously approved by NRC for the licensing calculations to operate the current Cycle 4 fuel. The safety MCPR limit of 1.06 is also used for analyses to support operation at conditions of final feedwater temperature reduction (FFTR) and is acceptable.

### 2.2 Operating MCPR Limits

Various transients could affect the operating MCPR limits during the extended Cycle 4 operation. The most limiting events (load rejection without bypass (LRWB) and feedwater controller failure (FWCF)) have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (RCPR). The RCPR values given in Table 3.3 of References 4 and 7 are cycle specific values for the Cycles 3 and 4 fuels using the COTRANSA (Ref. 5) and

XCOBRA-T (Ref. 6) methods which were previously approved by NRC. The operating MCPR values are determined by adding the RCPRs to the safety limit MCPR. The maximum MCPR values for different types of fuel (resulting from limiting event, LRWB) are specified as the operating MCPR limits and are incorporated into Table 3.2.3.1 of the Technical Specifications. The operating MCPR limits as a function of the core flow were also calculated and tabulated in Table 2.1 of Reference 3. Since the approved method was used to determine the operating MCPR limits to avoid violation of the safety limit MCPR in the event of any anticipated transients, we conclude that these limits are acceptable for operation of the extended Cycle 4 fuel.

Based on the review of the licensee's calculations in References 4 and 7 for fuels of the Cycles 3 and 4, we have also determined that the largest calculated CPR changes due to the effect of FFTR are acceptable for future extended reload applications with operation of FFTR provided that the assumptions made in performing transient analysis remain consistent with that in Reference 4. The acceptable limiting CPR changes for the LRWR and the FWCF events are 0.02 and -0.01 for the ANF fuel, and 0.01 and -0.01 for the GE fuel, respectively.

The application for the core reload amendment for the fifth fuel cycle include a report titled, "WNP-2 Cycle 5 Plant Transient Analysis," Revision 1, in which the licensee advised that based on analysis performed for final feedwater temperature reduction for Cycles 3 and 4 as well as Cycle 5 that "delta CPR changes for the LNRB and FWCF transients are conservatively bounded by adding 0.02 to the delta CPR values for these transients at normal feedwater temperature." By their supplemental letter dated June 1, 1989, (Ref. 12) they proposed these more conservative limits. We conclude that the limits are acceptable.

## 2.3 Thermal-Hydraulic Stability

The licensee has performed stability studies for the two limiting power/flow conditions (65% power and 45% flow, and 48% power and 27.6% flow). The calculated decay ratios are 0.52 and 0.49 for cases with and without feedwater temperature reduction at condition of 65% power and 45% flow. For 48% power and 27.6% flow case, the calculated decay decreases from 0.84 to 0.70 by inclusion of the effect of reduction in feedwater temperature. Since the calculated stability ratios for operation by the use of FFTR are bounded by that of cases with the normal feedwater temperature, the staff concludes that the results of the thermal-hydraulic stability analysis are acceptable for the extended fuel Cycle 4 operation with reduction in feedwater temperatures by as much as 65°F.

## 2.4 Reactor Vessel BeltLine Materials

Reactor vessel beltline materials are embrittled when they are irradiated by neutrons from the core. The procedures for calculating the amount of neutron radiation embrittlement of reactor vessel materials are contained in revision 2 to Regulatory Guide (RG) 1.99. Section 1.3 of the guide indicates that the procedures in the guide are applicable for irradiation temperatures between 525°F and 590°F. The licensee indicated that the minimum water temperature adjacent to the reactor vessel beltline was equivalent to the core inlet temperature, which for WNP-2 during normal power operation was 526.5°F to 533°F. The licensee indicated that the FFTR and thermal coastdown would not reduce the water temperature in the beltline region below 525°F. Hence, the reduction in feedwater temperature would not significantly change the amount of neutron radiation embrittlement and Revision 2 to RG 1.99 may be used to calculate the amount of neutron radiation embrittlement to WNP-2 reactor vessel beltline materials.

## 2.5 Reactor Vessel Feedwater Nozzles

BWR reactor vessel feedwater nozzles have experienced cracking in the bore and inner radius. The cracking was caused by high-cycle and low-cycle thermal fatigue stresses. The low-cycle thermal fatigue stresses result from system transients. FFTR will not affect this type of stress. However, high-cycle thermal fatigue stresses will be affected by the FFTR since these stresses result from turbulent mixing of hot reactor water (approximately 545°F) and the incoming feedwater. Since the FFTR will increase the difference in temperature between the hot reactor water and the feedwater the high-cycle thermal fatigue stress will increase.

The licensee determined the increase in high-cycle thermal fatigue stresses using the rapid cycle duty maps documented by General Electric in topical report NEDE-21821-A (Supplement 2, February 1980). The staff reviewed the General Electric method of calculating high-cycle thermal stresses in NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", November 1980. The staff concluded that the methods used and the results were acceptable.

The licensee's analysis indicates that the FFTR increased the high-cycle 40-year usage factor from 0.2047 to 0.2796 and the sum of the high-cycle and low-cycle usage factors for 40 years of operation was below the ASME Code limit of 1.0.

The increase in usage factor assumed that there was no leakage of cold feedwater through the thermal sleeve or its welds. If leakage through cracks in either the thermal sleeve or its welds occurred, the licensees analysis would be non-conservative and it is possible that cracks could initiate and propagate in the feedwater nozzle.

The thermal sleeve and its weld were fabricated using austenitic stainless steel material. Intergranular stress corrosion cracks (IGSCC) have been observed in austenitic stainless steel materials in BWR reactor water environnments. NUREG-0313, Rev. 2, January 1900 contains staff recommendations, which should prevent IGSCC of welded austenitic stainless steel material in BWR reactor water environment.

To evaluate whether the thermal sleeve would be susceptible to IGSCC, the staff requested that the licensee identify the materials used to attach the thermal sleeve to the nozzle safe-end. The thermal sleeve to thermal sleeve extension welds and thermal sleeve extension to nozzle safe-end welds were fabricated using Inconel 182 material. The thermal sleeves were fabricated using 304 stainless steel material with low carbon (.019%). The thermal sleeve extensions were fabricated from Inconel 600. Based on the recommendations in NUREG-0313, Rev. 2, all these materials, except for the Inconel 182 weld metals should not be susceptible to IGSCC. NUREG-0313, Rev. 2, indicates that Inconel 82 is the only commonly used nickel base weld considered to be resistant to IGSCC.

Since Inconel 182, a nickel base weld metal, was utilized in the thermal sleeve extension welds, the weld must be considered susceptible to IGSCC and leakage. Table 4-31 in NEDE 21821-A, a GE Tropical Report on this subject, indicates that if leakage increases from 0 GPM to 1.0 GPM through a thermal sleeve, the high cycle fatigue usage factor will increase by a factor of 6 and 225 for 420°F and 340°F rated feedwater temperature, respectively. This indicates that reducing the FFTR to 355°F and small amounts of leakage of feedwater through a cracked thermal sleeve during thermal coastdown could substantially increase the fatigue usage factor for the feedwater nozzle. Hence, leakage through cracks in the Inconel 182 weld metal and reducing the FFTR to 355°F could affect the integrity of the feedwater nozzle.

By letter dated February 14, 1990 (Ref. 14) the licensee indicated that the feedwater nozzle inspection program would be revised to schedule a volumetric inspection (ultransonic technique) of at least one nozzle during the refueling outage commencing after each use of feedwater temperature reduction to extend the fuel cycle. With this augmented inspection program to ensure that leakage through the Inconel 182 weld metal in the feedwater thermal sleeve extension welds does not result in cracking in the feedwater nozzle, the potential impact to the feedwater nozzles is acceptable.

## 2.6 Reactor Internals Load Impact

All the reactor internals (e.g., core plate, shroud support, shroud, top guide, shroud head, steam dryer, control rod guide tube, control rod drive housing and jet pump) were evaluated under the consideration of additional loads imposed by the ICF and FFWTR operations. Based on the duced in those components are within the ASME Code, Section III, Subsection NG allowables and the criteria referenced in the FSAR. The staff finds this acceptable since the original design criteria were satisfied.

## 2.7 Flow-Induced Vibration

To ensure that the flow-induced vibration response of the reactor internals for plant operation with ICF up to 106% rated flow is acceptable, the prototype (Tokai 2) plant test data as well as data from WNP-2 testing up to 106% core flow were reviewed. WNP-2 reactor internals were tested in accordance with provisions of Regulatory Guide 1.20, Revision 2 for non-prototype, Category IV plants using Tokai-2 as the limited valid prototype. This was described in the FSAR and was previously accepted by the staff. Based on the results of the licensee's assessment, the maximum alternating stress intensity for core flow up to 106% is 92% of the FSAR reactor internal vibration acceptance criterion. Therefore, the operation of WNP-2 at 106% rated core flow will not result in unacceptable reactor internal vibration.

### 2.8 Feedwater Nozzle and Feedwater Sparger Fatique Usage

At the end of the 1970's, inspections at 22 boiling water reactor plants identified cracking in the feedwater nozzle and sparger at 18 reactor vessels. The staff studied the issue and recommended hardware modifications, analysis methods and inspection schedules for the nozzle and sparger in NUREG-0619. The proposed FFWTR requested by the licensee will affect the fatique usage of the feedwater nozzle and sparger. The licensee indicated that the increase of the fatique usage factor due to feedwater temperature reduction was calculated using the analysis method described in GE report NEDE-21821-A (Supplement 2, February 1980). This GE report was referred to in NUREG-0619 and was approved by the staff in a letter from D. G. Eisenhut to R. G. Gridley dated January 14, 1980. Based on the result of the licensee's analysis, the 40-year total fatique usage factor with the proposed FFWTR operation remains below the ASME Code limit of 1.0 and is thus considered acceptable.

## 3.0 Technical Specification Changes

The following changes to the Technical Specifications and Bases, have been proposed for operation to extend the fuel cycle by reducing feedwater temperatures.

### 3.1 Technical Specification Sections 1.12A and 1.13A

One definition is added to each of the sections. The definitions are for the End-of-Cycle (EOC) and final feedwater temperature reduction, respectively. The changes are necessary to specify the conditions for the extended fuel cycle operation and are acceptable.

### 3.2 Technical Specifications Section 3/4.1.6

This is a new section, the limiting condition of operation (LCO) and surveillance requirements are added for the extended cycle operation with feedwater temperature reduction below 355°F. The changes are consistent with the current Technical Specifications with feedwater temperature reduced below the normal feedwater temperature of 420°F and are acceptable.

### 3.3 Technical Specifications Table 3.2.3.1

The operating MCPR limits based on the measured control rod scram times as well as the MCPR limits based on the control rod insertion time bounded by the Technical Specification limits are the GE and ANF fuels, during the fuel cycle extension operation. The changes are consistent with analytical results and are acceptable.

3.4 Figure 3.2.3-1

The note below the figure is revised to remove the phrase "when approved" since the issuance of this amendment constitutes approval of final feedwater temperature reduction. The curve is supported by analytical results and is acceptable.

3.5 Technical Specifications Basis 3/4.1.6.

A section is added to specify the feedwater temperature during the extended fuel cycle operation. The change is consistent with the assumptions used for the analysis to support the request of the Technical Specification changes and is acceptable.

3.6 Technical Specifications Basis 3/4.2-3

The statement is added to this section, which identifies the load rejection without bypass as the limiting case used to determine the operating MCPR limits for the extended fuel cycle. This change is supported by the analytical results discussed in this evaluation and is acceptable.

## 4.0 NO SIGNIFICANT HAZARDS DETERMINATION

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

The proposed amendment does not represent a significant hazard because it does not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The amendment allows operation with reduced feedwater temperature at the end of the fuel cycle to extend the length of the fuel cycle. The licensee analyzed the safety limits for the existing and reload fuel for the last three reloads and established safety limits to be applicable for the period during which feedwater temperature is reduced. The staff review of the licensee's submittal concluded that the analyses performed by the licensee and the limits established are acceptable. The licensee also analyzed the impact of the proposed operation with reduced feedwater temperature on reactor vessel beltline material,

on thermal stresses to feedwater nozzles, on mechanical stresses to reactor internals, on flow induced vibrations, and on material failure due to feedwater nozzle and feedwater sparger fatigue usage. The staff review found no significant change in previously accepted conditions.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve installation of new or different components. The amendment is limited to changes in operating procedures. Furthermore, the operational limits applied to WNP-2 have not been changed. Values for these limits to be applied during final feedwater temperature reduction were derived from NRC qualified computer codes and by applying the most limiting transients throughout the cycle. The review of the proposed limits and of the impact on mechanical components and structures did not identify the potential for any new or different kind of accident.

3) Involve a significant reduction in a margin of safety. The licensee's analysis of safety limits to be applicable during periods of reduced feedwater temperature resulted in a margin of safety either identical to or more conservative than that now used for the plant. The staff found the methods used by the licensee acceptable. The staff also completed its review of potential impacts to mechanical structures due to changes in core flow during feedwater temperature reduction period and metallurgical stresses due to reduced feedwater temperature and found these impacts acceptable. Therefore the staff concludes the change does not involve a significant reduction in a margin of safety.

For these reasons the staff has determined that the proposed amendment involves no significant hazards consideration.

## 5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in a requirement with respect to the installation and use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has determined that this amendment involves no significant hazards consideration. 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 published a notice of consideration of issuance of amendment to facility license and opportunity for hearing (53 FR 7270, March 7, 1988). No request for hearing or petition for leave to intervene was received. By letter dated March 31, 1989, the State of Washington advised that they do not have any comment.

## 7.0 CONCLUSION

In our review of the proposed changes to the Technical Specifications for operation of Cycle 4 with final feedwater temperature reduction, we find that approved methods have been used, and the results of the feedwater temperature reduction analysis support the proposed operating MCPR limits, which avoid violation of the safety MCPR limit for design transients. Therefore, the staff concludes that this core design will not adversely affect the capability to operate the plant safely during the extended fuel Cycle 4 operation at lower feedwater temperatures and the proposed Technical Specification changes are acceptable. The calculated CPR changes due to the effect of FFTR are also acceptable for the future reload applications provided that the assumptions used in performing transient analysis remain consistent with the assumptions used in Reference 4.

The method of calculating neutron radiation damage to reactor vessel beltline materials documented in revision 2 to RG 1.99 will be applicable for WNP-2 when the FFTR temperature is reduced from 420°F to 355°F and during thermal coastdown.

Since the materials used to fabricate the thermal sleeves and thermal sleeve extensions are not susceptible to IGSCC, leakage of cold feedwater through the thermal sleeve is not likely. Hence, the method used by the licensee to calculate high-cycle thermal fatigue stresses is acceptable and the usage factor for the feedwater nozzles during the 40 year life of the Plant should not exceed the ASME Code limit of 1.0.

The licensee has provided an adequate justification with respect to load impact of reactor internals, flow-induced vibrations as well as feedwater nozzle and sparger fatigue usage factor for the operation of WNP-2 with FFWTR and ICF up to 106% rated flow.

Based on the above conclusions, FFTR and thermal coastdown will not affect reactor vessel material integrity and the requested changes in technical specifications should be approved.

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.

## 8.0 REFERENCES


1. Letter from G. Sorensen (WPSS) to NRC, dated December 15, 1987 (G02-87-286).

2. NEDC-31107, Safety Review of WPPSS Nuclear Project No. 2 at Core Flow Conditions Above Rated Flow Throughout Cycle 1 and Fuel Feedwater Temperature Reductions, March 1986.

3. WPPSS-EANF-111, Final Feedwater Temperature Reduction Summary Report, November 1987.

4. XN-NF-87-92, WNP-2 Plant Transient Analysis with Final Feedwater Temperature Reduction, June 1987.

5. XN-NF-79-71, Revision 2 (as supplemented), Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, February 1987.

6. XN-NF-105(A), XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Code Analysis, February 1987.

7. Letter from G. Sorensen (WPPSS) to NRC, dated September 14, 1988 (G02-88-198) and XN-NF-87-92 (Supplement 1), WNP-2 Plant Transient Analysis with Final Feedwater Temperature Reduction Cycle 4 Analysis, May 1988.

8. Letter from G. Sorensen (WPPSS) to NRC, dated March 3, 1989 (G02-89-029).

9. Letter from G. Sorensen (WPPSS) to NRC, dated March 7, 1988 (G02-88-054).

10. Letter from G. Sorensen (WPPSS) to NRC, dated April 12, 1988 (G02-88-087).

11. Letter from G. Sorensen (WPPSS) to NRC, dated April 20, 1989 (G02-89-067).

12. Letter from G. Sorensen (WPPSS) to NRC, dated June 1, 1989 (G02-89-102).

13. Letter from G. Sorensen (WPPSS) to NRC dated February 14, 1990 (G02-90-024).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Dated: March 1, 1990

UNITED STATES NUCLEAR REGULATORY COMMISSION

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No.77 to Facility Operating License No. NPF-21, issued to Washington Public Power Supply System (the licensee), which revised the Technical Specifications for operation of the Nuclear Project No. 2, located in Benton County, Washington.

The amendment was effective as of the date of issuance.

This amendment adds a new section 3/4.1.6, "Reactivity Control Systems, Feedwater Temperature" which specifies that feedwater temperature shall not be reduced below 355°F. The amendment revised the MCPR Operating Limits in Table 3.2.3-1 by adding limits which would apply at the end of the fuel cycle when feedwater temperature is to be reduced. The amendment also adds definitions and revised the bases to cover feedwater temperature reduction.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter 1, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on March 7, 1988 (53 FR 7270). No request for a hearing or petition for leave to intervene was filed following this notice.

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.

For further details with respect to the action see (1) the application for amendment dated December 15, 1987 as supplemented by letters dated March 7, 1989, June 1, 1989, and February 14, 1990, (2) Amendment No. 77 to License No. NPF-21, (3) the Commission's related Safety Evaluation and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street NW., Washington, DC 20555, and at the Richland City Library, Swift and Northgate Streets, Richland, Washington 99352. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects III, IV, V and Special Projects.

Dated at Rockville, Maryland this 1st day of March, 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert B. Samworth

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