December 4, 1990

Docket No. 50-397

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. 90 TO FACILITY OPERATING LICENSE NO. NPF-21 FOR THE WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 74846)

The Commission has issued the enclosed Amendment No. 90 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 September 14, 1989 (G02-89-161).

This amendment revises Technical Specification Section 3/4 3.1 surveillance test intervals (STIs) and allowable outage times (AOTs) to reflect those stated in the Boiling Water Reactor Owner's Group (BWROG) Topical Report NEDC-30851P.

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

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

Enclosures:
1. Amendment No. 90 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)

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(10)

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. 90
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 September 14, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the 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. 90 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, 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: December 4, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 90

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
3/4 3-1	3/4 3-1
3/4 3-5	3/4 3-5
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
B3/4 3-1	B3/4 3-1

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within twelve hours. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

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*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within six hours after the channel was first determined to be inoperable or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	5(b)	3	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3	3
2. Average Power Range Monitor(c):			
a. Neutron Flux - High, Setdown	2	2	1
	3	2	2
	5(b)	2	3
b. Flow Biased Simulated Thermal Power - High	1	2	4
c. Fixed Neutron Flux - High	1	2	4
d. Inoperative	1, 2	2	1
	3	2	2
	5	2	3
3. Reactor Vessel Steam Dome Pressure - High	1, 2(e)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(d)	4	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to six hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations are being performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function shall be automatically bypassed when the reactor mode switch is not in the Run position and reactor pressure < 1037 psig.

(e) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 165 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

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*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	N.A.
b. Inoperative	N.A.
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	N.A.
b. Flow Biased Simulated Thermal Power - Upscale	6±1**
c. Fixed Neutron Flux - Upscale	≤ 0.09
d. Inoperative	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. Main Steam Line Radiation - High	N.A.
7. Primary Containment Pressure - High	N.A.
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	N.A.
b. Float Switch	N.A.
9. Turbine Throttle Valve - Closure	≤ 0.06
10. Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08#
11. Reactor Mode Switch Shutdown Position	N.A.
12. Manual Scram	N.A.

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U(b), S	S/U(c), W	R	2
	S	W	R	3, 4, 5
b. Inoperative	N.A.	W	N.A.	2, 3, 4, 5
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale, Setdown	S/U(b), S	S/U(c), W	SA	2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power - Upscale	S,D(g)	S/U(c), Q	W(d)(e), SA, R(h)	1
c. Fixed Neutron Flux - Upscale	S	S/U(c), Q	W(d), SA	1
d. Inoperative	N.A.	Q	N.A.	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	S	Q	R	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1, 2(i)
7. Primary Containment Pressure - High	N.A.	Q	R	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	N.A.	Q	R	1, 2, 5(j)
b. Float switch	N.A.	Q	R	1, 2, 5(j)
9. Turbine Throttle Valve - Closure	N.A.	Q	R	1
10. Turbine Governor Valve Fast Closure Valve Trip System Oil Pressure - Low	N.A.	Q	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	W	N.A.	1, 2, 3, 4, 5

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC 30851 P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the SER (letter to T. A. Pickens from A. Thadani dated July 15, 1987). The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C.-operated valves, a 3-second delay is assumed before the valve starts to move. For A.C.-operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C.-operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 13-second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints, and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

As part of the BWR Owners Group Technical Specification Improvement Program, General Electric (GE) performed a reliability analyses to identify improvements to the Reactor Protection System (RPS) surveillance test intervals (STI) and allowed outage times (AOT) as provided in the topical report, NEDC-30851P. NRC found that it provided an acceptable generic basis for supporting plant-specific technical specification (TS) changes related to the RPS. As noted in the SE for the GE topical report dated May 27, 1987, GE determined that if the proposed increase of RPS TS changes are implemented, there would be no significant increase of RPS failure frequency for the reviewed BWR plants. This determination is based on use of the GE procedure given in Appendix K of NEDC-30851P for evaluating specific plants against the generic RPS design and analyses.

The GE report does not confirm that calibration of the analog trip units can be extended from monthly to quarterly without creating excessive drift. As a result, the staff has identified plant-specific conditions that applicants must meet for proposed TS changes:

(1) Confirm the applicability of the generic analyses NEDC-30851P to its plant.

(2) Demonstrate, by use of current drift information provided by the equipment vendor or plant-specific data, that the drift characteristics for instrumentation used in the RPS channels in the plant are bounded by the assumption used in NEDC-30851P when functional test interval is extended from monthly to quarterly.

(3) Confirm that the difference between the parts of the RPS that perform the trip functions in the plant and those of the case plant were included in the analysis using the procedures of Appendix K of NEDC-30851P, or provide plant specific analyses to demonstrate that there is no appreciable change in RPS availability or public risk.

In accordance with the plant-specific conditions that each licensee must meet to make any proposed Technical Specification changes fully acceptable, the Washington Public Power Supply System (WPPSS) proposed changes to the TS related to the RPS for the Washington Nuclear Project Unit 2 (WNP-2) by letter dated September 14, 1989.

2.0 EVALUATION

The generic study in NEDC-30851P provided a technical basis to modify the surveillance test frequencies and allowable out-of-service times for RPS components from the generic TS. The generic study also provided additional analyses of various known RPS configurations to support the applications of the generic basis on the plant-specific basis. In the submittal of September 14, 1989, the licensee stated that the generic study applies to WNP-2.

The staff SE of May 27, 1987 on GE Topical Reports NEDC-30844 and NEDC-30851P states that licensees should examine plant and/or generic data from representative instrument channels over a sufficient period to demonstrate that the setpoint drift expected with the extended STIs is within the margins established using their current methodology. By letter to BWR Owners Group from C. Rossi (NRC) dated April 27, 1988, the NRC requested licensees to confirm that the setpoint drift which could be expected under the extended STIs had been studied and either (1) had been shown to remain within the existing allowance in the RPS and engineered safety features instrument setpoint calculations or (2) that allowances and setpoints had been adjusted to account for the additional expected drift. No additional information needed to be provided for staff review. However, records showing the actual setpoint calculation and supporting data should be retained onsite for possible future staff audit.

The licensee stated that they had examined plant specific setpoint drift data and characteristics of the subject RPS equipment, and confirmed that the setpoints will not drift beyond the existing allowance during the quarterly surveillance test interval.

The licensee also stated that plant specific characteristics of its reactor protection system do not appreciably differ from the generic base case plant in either RPS availability or public risk, and referred to a WNP-2 specific evaluation of modifying the surveillance test frequencies and allowable out-of-service time of the RPS from the existing TS performed by GE. In addition, the licensee stated that these TS changes do not degrade RPS function or reliability. Therefore, the licensee states that no unanalyzed mode of operation or kind of accident results from these changes.

The licensee requests an amendment to Technical Specification 3/4.3.1. The following changes are proposed:

(1) Extending weekly and monthly channel function test frequencies to quarterly (except for the Manual function, which was changed from monthly to weekly). This change applies to Table 4.3.1.1-1, REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS, Functional Units 2.b, c, and d, 3, 4, 5, 6, 7, 8a, 9, 10, and 12.

(2) Extend AOTs for the repair of one trip system from 1 to 12 hours. This

change applies to Specification 3.3.1: Replace the words "1 hour" in Action a. with the words "twelve hours".

(3) Extend AOTs for channel surveillance testing and for repair from 2 to 6 hours when both trip systems are potentially degraded. This change applies to Table Notation (a) of Table 3.3.1-1.

(4) Replace the words "2 hours" in the footnote to Action a. of Specification 3.3.1 with the words "6 hours after the channel was first determined to be inoperative."

Based on a review of the licensee's submittal, we find the four proposed changes acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, or changes a surveillance requirement. The 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 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 the amendment.

4.0 CONCLUSION

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and 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 Contributors: P. Loeser
P. Eng

Dated: December 4, 1990