June 26, 1995

Mr. J. V. Parrish (Mail Drop 1023)
Vice President Nuclear Operations
3000 George Washington Way
Washington Public Power Supply System
P.O. Box 968
Richland, Washington 99352-0968

SUBJECT: ISSUANCE OF AMENDMENT FOR THE WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2 (TAC NO. M89907)

Dear Mr. Parrish:

The Commission has issued the enclosed Amendment No. 139 to the Facility Operating License No. NPF-21 for WPPSS Nuclear Project No. 2. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated July 12, 1994.

The amendment modifies the technical specifications (TS) to remove instrument response time limit tables for the reactor protection system (RPS), isolation actuation, and emergency core cooling system (ECCS) from the TS. The affected instrument response time limit tables will be located in the Final Safety Analysis Report (FSAR).

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
James W. Clifford, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. Amendment No. 139 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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DOCUMENT NAME: WNP89907.AMD

OFC	LA:PDIV-2	PDIV-2	SRXB	HICB	OGC
NAME	EPeyton	JClifford	RJones for	JWermiel	RBachmann
DATE	5/9/95	5/31/95	6/6/95	6/7/95	6/16/95

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WBateman	RIV, WCFO (4)
LHurley, RIV DRS	
RJones	
JWermiel	

DOCUMENT NAME: WNP89907.AMD

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DATE	5/9/95	5/31/95	6/6/95	6/7/95	6/16/95

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 26, 1995

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Vice President Nuclear Operations
3000 George Washington Way
Washington Public Power Supply System
P.O. Box 968
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James W. Clifford, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. Amendment No. 139 to NPF-21
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
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 (licensee) dated July 12, 1994, 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. 139 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 its date of issuance to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James W. Clifford, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 26, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 139 TO 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	INSERT
xxi	xxi
3/4 3-1	3/4 3-1
3/4 3-6	3/4 3-6
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-12*	---
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-25	3/4 3-25
3/4 3-26*	---
3/4 3-33	3/4 3-33
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2	B 3/4 3-2

*These pages are unchanged. They are being reissued to retain consistency as being overleaf pages.

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LIST OF TABLES

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.

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 shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. 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.

---

* 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 < 1060 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 based on turbine first stage pressure when THERMAL POWER is less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

---

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

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TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decade during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Measure and compare core flow to rated core flow.

(h) This calibration shall consist of verifying the 6 $\pm$ 1 second simulated thermal power time constant.

(i) DELETED

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

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system:

1. If placing the inoperable channel(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to OPERABLE status within

a) 12 hours for trip functions common to RPS Instrumentation; and

b) 24 hours for trip functions not common to RPS Instrumentation.

or the ACTION required by Table 3.3.2-1 for the affected trip function shall be taken.

OR

2. If placing the inoperable channel(s) in the tripped conditions would not cause an isolation, the inoperable channel(s) and/or that trip system shall be placed in the tripped condition within

a) 12 hours for trip functions common to RPS Instrumentation; and

b) 24 hours for trip functions not common to RPS Instrumentation.

The provisions of Specification 3.0.4 are not applicable.

c. 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 one hour and take the ACTION required by Table 3.3.2-1.

---

*Place one trip system (with the most inoperable channels) in the tripped condition. The trip system need not be placed in the tripped condition when this would cause the isolation to occur.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation 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 4.3.2.1-1.

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

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shall be demonstrated to be within its limit at least once per 18 months. Radiation detectors are exempt from response time testing. 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 isolation trip system.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	5(g)	2	1, 2, 3	20
2) Low Low, Level 2	1, 2, 4	2	1, 2, 3	20
b. Drywell Pressure - High	4, 5(b)(g)	2	1, 2, 3	20
c. Main Steam Line				
1) DELETED				
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2(d)	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2	1, 2, 3	21
e. Main Steam Line Tunnel Δ Temperature - High	1	2	1, 2, 3	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
g. Manual Initiation	1	2/group	1, 2, 3	24
	2	1/group	1, 2, 3	24
	5(b)(g)	1/group	1, 2, 3	24
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	3(b)(e)	2	1, 2, 3, and **	25
b. Drywell Pressure - High	3(b)(e)	2	1, 2, 3	25
c. Reactor Vessel Water Level - Low Low, Level 2	3(b)(e)	2	1, 2, 3, and #	25
d. Manual Initiation	3(b)	1/group	1, 2, 3	24
	3(b)	1/group	**	24

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TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-				
1) Low, Level 3	S	Q	R	1, 2, 3
2) Low Low, Level 2	N.A.	Q	R	1, 2, 3
b. Drywell Pressure - High	N.A.	Q	R	1, 2, 3
c. Main Steam Line				
1) DELETED				
2) Pressure - Low	N.A.	Q	R	1
3) Flow - High	S	Q	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	N.A.	SA	R	1, 2, 3
e. Main Steam Line Tunnel Δ Temperature - High	N.A.	SA	R	1, 2, 3
f. Condenser Vacuum - Low	N.A.	Q	R	1, 2*, 3*
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	Q	R	1, 2, 3, and **
b. Drywell Pressure - High	N.A.	Q	R	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	N.A.	Q	R	1, 2, 3, and #
d. Manual Initiation	N.A.	R	N.A.	1, 2, 3, and **

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, within 24 hours take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status:

   1. Within 7 days, provided that the HPCS and RCIC systems are OPERABLE; otherwise,

   2. Within 72 hours.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 128 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation 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 4.3.3.1-1.

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

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shall be demonstrated to be within the 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 ECCS trip system.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
A. DIVISION I TRIP SYSTEM			
1. RHR-A (LPCI MODE) & LPCS SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. LPCS Pump Discharge Flow-Low (Minimum Flow)	1	1, 2, 3, 4*, 5*	31
d. Reactor Vessel Pressure-Low (LPCS Permissive)	1	1, 2, 3, 4*, 5*	32 33
e. Reactor Vessel Pressure-Low (LPCI Permissive)	1	1, 2, 3, 4*, 5*	32 33
f. LPCI Pump A Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32
g. LPCI Pump A Discharge Flow-Low (Minimum Flow)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2	1, 2, 3	30
b. ADS Timer	1	1, 2, 3	32
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
d. LPCS Pump Discharge Pressure-High (Pump Running)	2	1, 2, 3	32
e. LPCI Pump A Discharge Pressure-High (Pump Running)	2	1, 2, 3	32
f. Manual Initiation	2/division	1, 2, 3	35
g. Inhibit Switch	1/division	1, 2, 3	35

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TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	N.A.	Q	R	1, 2, 3
c. LPCS Pump Discharge Flow-Low (Minimum Flow)	N.A.	Q	R	1, 2, 3, 4*, 5*
d. Reactor Vessel Pressure-Low (LPCS Permissive)	N.A.	Q	R	1, 2, 3, 4*, 5*
e. Reactor Vessel Pressure-Low (LPCI Permissive)	N.A.	Q	R	1, 2, 3, 4*, 5*
f. LPCI Pump A Start Time Delay Relay	N.A.	Q	Q	1, 2, 3, 4*, 5*
g. LPCI Pump A Flow-Low (Minimum Flow)	N.A.	Q	R	1, 2, 3, 4*, 5*
h. Manual Initiation	N.A.	R	N.A.	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM				
TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3
b. ADS Timer	N.A.	Q	Q	1, 2, 3
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	S	Q	R	1, 2, 3
d. LPCS Pump Discharge Pressure-High (Pump Running)	N.A.	Q	R	1, 2, 3
e. LPCI Pump A Discharge Pressure-High (Pump Running)	N.A.	Q	R	1, 2, 3
f. Manual Initiation	N.A.	R	N.A.	1, 2, 3
g. Inhibit Switch	N.A.	Q	N.A.	1, 2, 3

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 RPS instrumentation that provides 1) the Turbine Throttle Valve-Closure and 2) Turbine Governor Valve Fast Closure, Valve Trip System Oil Pressure - Low trip signals measures first stage turbine pressure to initiate a trip signal. The Load Rejection safety analysis (FSAR 15.2.2) bases initial conditions on rated power and specifies turbine bypass operability at greater than or equal to 30% of rated thermal power. Because first stage pressure can vary depending on operating conditions, the qualifying notes describing when the turbine bypass feature is to be disabled specify a turbine first stage pressure corresponding to less than 30% RTP (turbine first stage pressure is dependent on the operating parameters of the reactor, turbine, and condenser). Therefore, because a value for turbine first stage pressure cannot be precisely fixed and because pressure measurement initiates the trip, the Technical Specification refers to a pressure associated with a specific Rated Thermal Power value rather than a value for pressure.

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. The response time limits are contained in FSAR Chapter 7.

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 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. The response time limits are contained in FSAR Chapter 7.

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 and trip setpoints 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. The response time limits are contained in FSAR Chapter 7.

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 139 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 July 12, 1994, the Washington Public Power Supply System (WPPSS or the licensee) proposed that Appendix A of Facility Operating License NPF-21 be amended to revise the WPPSS Nuclear Project No. 2 Technical Specifications (TS). The changes modify the action statements of TS 3.3.1, 3.3.2, and 3.3.3 to allow relocation of Tables 3.3.1-2, 3.3.2-3, and 3.3.3-3 from the TS to the Final Safety Analysis Report (FSAR). Additional changes to TS surveillance requirements 4.3.1.3 and 4.3.2.3 incorporate existing exclusions for neutron monitors and radiation detectors from response time testing requirements. The TS changes and relocated tables are associated with reactor protection system (RPS), isolation actuation system (IAS), and emergency core cooling system (ECCS) instrumentation.

The NRC provided guidance to all holders of operating licenses or construction permits for nuclear power reactors on these TS changes in Generic Letter (GL) 93-08, "Relocation of Technical Specification Tables of Instrument Response Time Limits," dated December 29, 1993.

2.0 BACKGROUND

In the early 1980s, the NRC staff undertook efforts to address problems related to the content of nuclear power plant TS. These projects have resulted in the issuance of various reports, proposed rulemakings, and Commission policy statements. Line item improvements became a mechanism for TS improvement as part of the implementation of the Commission's interim policy statement on TS improvements published on February 6, 1987 (52 FR 3788). The final Commission policy statement on TS improvements was published July 22, 1993 (58 FR 39132). The final policy statement provided criteria that the industry can use to more clearly establish the framework for TS. The NRC staff has maintained the line item improvement process, through the issuance of generic letters, to improve the content and consistency of TS and to reduce the licensee and staff resources required to process amendments related to the TS being relocated from the TS to other licensee documents as a result of the implementation of the Commission's final policy statement.

Section 50.36 of Title 10 of the Code of Federal Regulations establishes the regulatory requirements for licensees to include TS as part of the application for an operating license. The rule requires that TS include items in five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operations; (3) surveillance requirements; (4) design features; and (5) administrative controls. In addition, the Commission's final policy statement on TS improvements and other Commission documents provide guidance regarding the required content of TS. The fundamental purpose of the TS, as described in the Commission's final policy statement, is to impose those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety and to establish certain conditions of operation that cannot be changed without prior Commission approval.

The Commission's final policy statement recognized, as had previous statements related to the NRC staff's TS improvement program, that implementation of the policy would result in the relocation of existing TS requriements to licensee-controlled documents such as the FSAR. Those items relocated to the FSAR would in turn be controlled in accordance with the requirements of 10 CFR 50.59, "Changes, tests, and experiments." Section 50.59 provides criteria to determine when changes to a facility, procedures, or tests and experiments planned by a licensee require prior Commission approval in the form of a license amendment in order to address any unreviewed safety questions or changes to the TS. NRC inspection and enforcement programs also enable the NRC staff to monitor facility changes and licensee adherence to FSAR commitments and to take appropriate remedial action.

3.0 EVALUATION

The licensee proposed changes to TS 3.3.1, TS 3.3.2, and TS 3.3.3 that remove references to Tables 3.3.1-2, 3.3.2-3, and 3.3.3-3. The change also removes the listed tables from the TS. The licensee also proposed to incorporate existing exclusions for response time testing of neutron detectors and radiation monitors, identified in TS Surveillance Requirements 4.3.1.3 and 4.3.2.3, into the revised TS. In its July 12, 1994, amendment request, in accordance with the schedule requirements of 10 CFR 50.71, the licensee committed to include the relocated instrument response time limits in the next FSAR update after their request is approved.

Tables 3.3.1-2, 3.3.2-3, and 3.3.3-3 contain the required values for the response time limits for the RPS, IAS, and ECCS instruments. The limiting conditions for operation (LCOs) for the RPS, IAS, and ECCS instrumentation specify that these systems must be operable with the response times as specified in these tables. Action statements for systems that do not meet the operability requirements are contained in TS 3.3.1, TS 3.3.2, and TS 3.3.3. The response time limits are also the acceptance criteria for the response time tests performed to satisfy the surveillance requirements of TS 4.3.1.3, TS 4.3.2.3, and TS 4.3.3.3 for each applicable RPS, IAS, and ECCS trip function. The surveillance requirements ensure that the response times of the RPS, IAS, and ECCS instruments are consistent with the assumptions of the

safety analyses performed for design basis accidents and transients. The changes associated with the implementation of GL 93-08 involve only the relocation of the RPS, IAS, and ECCS response time limit tables to the FSAR. The changes do not affect the TS action statements for inoperable instrumentation, nor do they affect the surveillance requirements to perform response time testing. The FSAR will contain the acceptance criteria for the required RPS, IAS, and ECCS response time surveillances.

The NRC staff considers that the removal of the specific response time tables does not eliminate the requirements for the licensee to ensure that the protection instrumentation is capable of performing its intended safety functions. Relocation of the the specific values of the required response times does not change the licensee's responsibility to evaluate any changes to response time requirements in accordance with the requirements of 10 CFR 50.59. If the licensee wanted to change any of the response times in the relocated table, the licensee would have to determine whether the change involved an unreviewed safety question. If the licensee determined that any such proposed change involved either (1) an increase in the probability or consequences of accidents or malfunctions of equipment important to safety, (2) the creation of a possibility for an accident or malfunction of a different type than any evaluated previously, or (3) a reduction in a margin of safety, the licensee would have to obtain prior NRC approval of a license amendment before implementing the proposed change.

The NRC also determined that 10 CFR 50.36 does not require that the response time tables be retained in the TS. The TS will retain the requirements related to operability, applicability, and the surveillance requirements, including the requirement to conduct testing to ensure the response times for RPS, IAS, and ECCS are within applicable limits, because of the importance of these systems in mitigating the consequences of an accident. The NRC staff considers the response times themselves to be an operational detail related to the licensee's safety analyses, which are generally discussed in the FSAR and controlled by review of changes against the criteria of 10 CFR 50.59. The continued processing of license amendments related to modification of the affected instrument response times, where the revisions to those times do not involve an unreviewed safety question under 10 CFR 50.59, would afford no significant benefit with regard to protecting the public health and safety. Further, the response time limits do not constitute a condition or limitation on operation necessary to obviate the possibility of an abnormal situation or event posing an immediate threat to the public health and safety, since the ability of the RPS, IAS, and ECCS to perform their safety functions is not affected by the relocation of the response time tables from the TS to the FSAR.

The NRC staff concludes that the changes do not alter the TS requirements to ensure that the response times of the RPS, IAS, and ECCS instruments are within their limits. In addition, the TS changes are consistent with the guidance provided in GL 93-08 and the requirements of 10 CFR 50.36. On these bases, the NRC staff concludes that the relocation of these response time limit tables from the TS to the FSAR is acceptable.

4.0 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.

5.0 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 and changes surveillance requirements. 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 (59 FR 45036). 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: Jim Clifford

Date: June 26, 1995