January 6, 1989

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. 64 TO FACILITY OPERATING LICENSE NO. NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 71445)

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 21, 1988.

This amendment revises the testing requirements for 4.16 KV emergency bus undervoltage trip functions set forth in WNP-2 Technical Specification Tables 3.3.3-1, 3.3.3-2 and 4.3.3.1-1.

Because this amendment is needed to avoid the necessity to shut down the plant on January 8, 1989, it was authorized on an emergency basis.

A copy of the related safety evaluation is enclosed. The notice of issuance and final determination of no significant hazards consideration and opportunity for hearing will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

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. 64 to Facility Operating License No. NPF-21
2. Safety Evaluation

cc: w/enclosures
See next page

DF01
1/1
as modified

*PREVIOUSLY CONCURRED

DRSP/PD5	*DRSP/APM:PD5	*NRR/SELB	OGC	DRSP/D:PD5	(A)AD;DRSP35
JLee:dr	RSamworth	FRosa		GWKnighton	MVirgilio
1/5/88	12/29/88	12/30/88	1/5/88	1/5/88	1/6/89

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

January 6, 1989

Docket No. 50-397

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Regulatory Programs
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P.O. Box 968
3000 George Washington Way
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Because this amendment is needed to avoid the necessity to shut down the plant on January 8, 1989, it was authorized on an emergency basis.

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Sincerely,

for Robert B. Samworth, Senior Project Manager
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Enclosures:
1. Amendment No. 64 to Facility Operating License No. NPF-21
2. Safety Evaluation

cc: w/enclosures
See next page

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

WPPSS Nuclear Project No. 2
(WNP-2)

cc:
Nicholas S. Reynolds, Esq.
Bishop, Cook, Purcell
& Reynolds
1400 L Street NW
Washington, D.C. 20005-3502

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

Mr. Curtis Eschels, Chairman
Energy Facility Site Evaluation Council
Mail Stop PY-11
Olympia, Washington 98504

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

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

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

Mr. C. M. Powers
WNP-2 Plant Manager
Washington Public Power Supply System
P. O. Box MD 927M
Richland, Washington 99352

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

Chairman
Benton County Board of Commissioners
Prosser, Washington 99350

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. 64
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 21, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

    B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

    C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

    D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

    E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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. 64, 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

Harry Rood for GK

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III, IV, V and Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 6, 1989

ENCLOSURE TO LICENSE AMENDMENT NO. 64

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
3/4 3-28	3/4 3-27
3/4 3-32	3/4 3-31
3/4 3-36	3/4 3-35

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
B. DIVISION 2 TRIP SYSTEM			
1. RHR B and C (LPCI MODE)			
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. Reactor Vessel Pressure-Low (LPCI Permissive)	1/valve	1, 2, 3, 4*, 5*	32 33
d. LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32
e. LPCI Pump Discharge Flow-Low (Minimum Flow)	1/pump	1, 2, 3, 4*, 5*	31
f. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
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. LPCI Pump B and C Discharge Pressure - High (Pump Running)	2/pump	1, 2, 3	32
e. Manual Initiation	2/division	1, 2, 3	35
f. Inhibit Switch	1/division	1, 2, 3	35

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. Reactor Vessel Water Level-High, Level 8	2(c)	1, 2, 3, 4*, 5*	32
d. Condensate Storage Tanks Level-Low	2(d)	1, 2, 3, 4*, 5*	36
e. Suppression Pool Water Level-High	2(d)	1, 2, 3, 4*, 5*	36
f. HPCS System Flow Rate-Low (Minimum Flow)	1	1, 2, 3, 4*, 5*	31
g. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
D. LOSS OF POWER					
1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	2/bus	1/bus	2/bus	1, 2, 3, 4**, 5**	37
2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage Division 1 and 2)	3/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	38
3. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage Division 3)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	38

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also activates the associated division diesel generator.

(c) Provides signal to close HPCS pump discharge valve only on 2-out-of-2 logic.

(d) Provides signal to HPCS pump suction valves only.

* When the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

** Required when ESF equipment is required to be OPERABLE.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 128 psig.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
B. DIVISION 2 TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.65 psig	≤ 1.85 psig
c. Reactor Vessel Pressure-Low (LPCI Permissive)	≥ 470 psig, decreasing	≥ 450 psig, decreasing
d. LPCI Pump B Start Time Delay Relay	≤ 5 seconds	≤ 6 seconds
e. LPCI Pump Discharge Flow-Low (Minimum Flow)	≥ 800 gpm	≥ 650 gpm
f. Manual Initiation	N.A.	N.A.
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. ADS Timer	≤ 105 seconds	≤ 117 seconds
c. Reactor Vessel Water Level-Low, Level 3 (Permissive)	≥ 13.0 inches*	≥ 11 inches
d. LPCI Pump B and C Discharge Pressure-High (Pump Running)	≥ 125 psig, increasing	≥ 115 psig, increasing
e. Manual Initiation	N.A.	N.A.
f. Inhibit Switch	N.A.	N.A.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches
b. Drywell Pressure - High	≤ 1.65 psig	≤ 1.85 psig
c. Reactor Vessel Water Level - High, Level 8	≤ 54.5 inches*	≤ 56.0 inches
d. Condensate Storage Tank Level - Low	≥ 448 ft 3 in. elevation	≥ 448 ft 0 in. elevation
e. Suppression Pool Water Level - High	≤ 466 ft 8 in. elevation	≤ 466 ft 10 in. elevation
f. HPCS System Flow Rate - Low (Minimum Flow)	≥ 1250 gpm	≥ 1200 gpm
g. Manual Intiation	N.A.	N.A.
D. LOSS OF POWER		
1. 4.16 kV Emergency Bus Undervoltage Loss of Voltage ##	a. 4.16 kV Basis - 2870 ± 86 volts b. 120 V Basis - 82 ± 2.5 volts	2870 ± 172 volts 82 ± 5 volts
a. Divisions 1 and 2 b. Division 3	a. 4.16 kV Basis - 3016 ± 90 volts b. 120 V Basis - 87 ± 2.5 volts	3016 ± 180 volts 87 ± 5 volts
2. 4.16 kV Emergency Bus Undervoltage Degraded Voltage (Divisions 1, 2, and 3	a. 4.16 kV Basis - 3632 ± 108 volts b. 120 V Basis - 104.0 ± 3.0 volts c. 8 ± 0.4 sec time delay	3632 ± 216 volts 103.8 ± 6.0 volts 8 ± 0.8 sec time delay

TABLE NOTATIONS

*See Bases Figure B 3/4 3-1.

##These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	N.A.	M	R	1, 2, 3
c. Reactor Vessel Water Level-High, Level 8	S	M	R	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	N.A.	M	R	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	N.A.	M	R	1, 2, 3, 4*, 5*
f. HPCS System Flow Rate-Low (Minimum Flow)	N.A.	M	R	1, 2, 3, 4*, 5*
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	N.A.	R	1, 2, 3, 4**, 5**
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage Division 1 and 2)	N.A	M***	R	1, 2, 3, 4**, 5**
3. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage Division 3)	N.A.	N.A.	R	1, 2, 3, 4**, 5**

TABLE NOTATIONS

#Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 128 psig.

*When the system is required to be OPERABLE per Specification 3.5.2.

**Required when ESF equipment is required to be OPERABLE.

***The secondary time delay 3 second relays are exempt from this monthly testing. The secondary time delay relays associated with this logic will be functionally tested as part of the Logic System Functional Testing (Surveillance Requirement 4.3.3.2)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 64 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 December 21, 1988, Washington Public Power Supply System proposed certain changes to the Technical Specifications for Nuclear Project No. 2.

Specifically, the Supply System is requesting that Tables 3.3.3-1, 3.3.3-2 and 4.3.3.1-1 be revised to reduce the testing requirements and capabilities of the Divisions 1, 2 and 3 Loss of Power trip functions.

Table 4.3.3.1-1 Item D.2, Loss of Power, 4.16 KV Emergency Bus Undervoltage (Degraded Voltage) presently requires a channel functional test (CFT) monthly. This requirement and the notation on Table 3.3.3-2, Emergency Core Cooling System Actuation Instrumentation Setpoints, that the associated time delay relay (TDR) is 8± 0.04 second, dictates that the monthly CFT be done on circuitry encompassing an 8-second time delay. This circuitry is comprised of two sequential time delays. One delay (5 seconds) is in the circuitry sensing the degraded voltage condition and the other (3 seconds, the secondary TDR) provides circuit trips to obtain the next reliable source of power for the Emergency Core Cooling System (ECCS) equipment. The design of the secondary TDR precludes testing at power. Hence, the monthly CFT on the 3-second time delay relay would require shutdown to implement this testing. Table 4.3.3.1-1 would be revised to note that the secondary TDR is tested during Logic System Functional Testing and exempted from the monthly CFT.

Table 3.3.3-2 Item D, Loss of Power, lists relay tolerances for the trip setpoint TDR for the degraded voltage setpoint as ±0.04. This value is thought to be a transcription or typing error such that the setpoint tolerance should be ±0.4 seconds instead of the listed 0.04. It is proposed to change this tolerance to ±0.4.

Tables 3.3.3-1 and 4.3.3.1-1 presently require the same testing of Divisions 1, 2 and 3. The Division 3 design is such that the monthly CFT required by Table 4.3.3.1-1 implies plant shutdown. The proposed change to Table 4.3.3.1-1 would require the calibration of Division 3 on an 18-month schedule with no monthly testing.

The licensee made the request for this license amendment on an emergency basis, stating that without the amendment it will be necessary to shut down on January 8, 1989 and monthly thereafter to do the surveillance on the degraded voltage protection instrumentation.

2.0 EVALUATION

The objective of monthly testing of emergency core cooling system actuation instrumentation is to achieve and ensure an acceptable level of reliability for the instrumentation. Recent review of the surveillance requirements in the WNP-2 technical specifications has identified that testing requirements imply capabilities that are not inherent in the instrumentation design. System design and the quality of installed components contribute to the reliability of the instrumentation. This evaluation addresses the amount of testing necessary to ensure reliability.

Division 1 and 2 Degraded Voltage Instrumentation

The trip setpoints include the voltage at which the emergency buses would disconnect from the offsite power and the time period that the degraded voltage can continue before separation occurs. This amendment request pertains specifically to time delay setpoint and to the capability of the protection system to perform its function of separating the emergency buses from the offsite power.

The original design of the degraded voltage protection considered the allowable drift, setting tolerances and repeatability to select the appropriate device to perform the function.

The 3-second time delay relays do not have a high accuracy requirement imposed on them in the present degraded voltage transfer scheme at WNP-2. In the overall ECCS actuation timing as described in Table 6.3-1 of the WNP-2 FSAR, plus or minus 10% can be allowed on this time delay to cover both accuracy and drift and still retain a large margin to the protective limit assigned to reenergizing the Division 1 and 2 buses from the onsite power sources. The total time assigned to energizing these buses from the onsite power source is approximately 19 seconds. If degraded voltage is sensed, the nominal time (total) to complete the transfer to the backup offsite power source is 10 seconds and 13 seconds to energize the bus from the diesel generator source. Hence this scheme provides 9 seconds gross margin for the transfer to the backup source and 6 seconds margin for the transfer to the diesel generator source without impinging on the other margins in the ECCS actuation assumptions. The allowance of

+20% to cover setability, drift, and accuracy for these 3-second relays is conservative and only reduces the overall margin by 0.6 seconds. The drift on these relays on the long side could exceed 100% and still permit the system to meet the ECCS actuation assumption for a design basis assumption of a LOCA and concurrent degraded network voltage.

The allowance of -20% in the short time direction allows the scheme to ride through a combination of a worst case motor start (HPCS) and a degraded network source that produces a voltage of 80% on the Division 1 and 2 buses. If the network is degraded more, the system will initiate a transfer of the Division 1 and 2 buses to the next available source. If the 3-second relay drifts beyond the allowance in the short time direction, it only means that the scheme will not ride through as much degradation (both voltage level and time) before it initiates a transfer. There will be no loss of the transfer function.

Because the allowable time for separation from the degraded voltage source has a wide margin, and because this margin is not significantly reduced by the proposed less frequent testing of the secondary TDR, the staff agrees that monthly testing of this TDR is not necessary. The proposed amendment for testing of the secondary TDR is acceptable.

Table 3.3.3-2 Item D, Loss of Power, lists relay tolerances for the trip setpoint TRD for the degraded voltage setpoint as +/-.04. This accuracy is not attainable and the entry in the Technical Specifications is thought to be a transcription or typing error such that the setpoint tolerance should be +/-.4 seconds instead of the listed .04. A value of .4 was also derived using WNP-2 setpoint methodology recently approved by the NRC. Additionally, the .4 value is consistent with Standard Technical Specification notation showing +/-5% for setpoint tolerances. The staff agrees that the use of values consistent both with NRC approved setpoint methodology and the Standard Technical Specification criteria is appropriate and finds the change acceptable.

Division 3 Degraded Voltage Instrumentation

Tables 3.3.3-1 and 4.3.3.1-1 presently do not differentiate between Division 1 and 2 and Division 3 requirements. Division 3 is committed to serving the high pressure core spray (HPCS) system. The Division 3 design is not reflected accurately in the Table 3.3.1-1 channel descriptions nor are testing capabilities reflected accurately in Table 4.3.3.1-1. The monthly CFT required by Table 4.3.3.1-1 for Division 3 would require plant shutdown for testing.

The testing frequency for the degraded voltage sensing relays of the Division 3 emergency bus is proposed to be changed to require annual testing as part of the Logic System Functional Testing.

The design of the Division 3 degraded voltage protection is unique in that two relays monitor the offsite source and initiate a transfer to the emergency diesel generator upon sensing a degraded voltage condition for an extended time. There is no backup offsite source available to the Division 3 bus. The Division 3 bus is dedicated solely to the HPCS system.

The degraded voltage protection relays are ITE 27N (the same model utilized for the sensor relays in Divisions 1 and 2) and are high precision relays with excellent repeatability. Vendor information provided indicates long term stability (drift) is expected to be ±.2 volt per year. The licensee's experience with the annual calibration check of these relays has been that yearly drift has been small. It is not anticipated that the time delay drift associated with these devices will be significant. Short term drift of these devices on the Division 1 and 2 buses will be monitored during the monthly Channel Functional Tests. Any significant drift in voltage or time delay setpoints noted on the Division 1 and 2 relays will initiate appropriate action to assure Division 3 operability.

The allowable margins for the time delay specification discussed above for Division 1 and 2 can readily be met with these devices. Design modifications to the Division 3 circuit, which may improve reliability and reduce risk by allowing more frequent testing, are included but are not required to provide adequate protection for the public health and safety.

The staff finds the proposed amendment to specify testing of these devices at a maximum frequency of 18 months acceptable.

3.0 EMERGENCY CIRCUMSTANCES

On November 18, 1988, the licensee discovered and reported (see LER 88-036 dated December 19, 1988) that a surveillance of the TDR, normally performed as part of the Channel Calibrations while in refueling outages, had been missed during the Spring 1988 outage. Based on discussions with the NRC staff prior to plant licensing, the licensee had interpreted the Technical Specifications such that testing of the 3-second delay relay was not required monthly by the Technical Specifications. Numerous formal and informal discussions between the Supply System and the NRC Staff in November and December of 1988 (see WPPSS letter dated December 7, 1988 and NRC Summary of December 15, 1988 Meeting dated January 5, 1989) led to a correct interpretation of the specific Technical Specification requirements such that, as currently stated, the surveillance testing is required monthly even though the testing cannot be accomplished at power because the testing procedure itself will cause a reactor trip. During these discussions the licensee agreed to submit and justify a request for a Technical Specification change that would modify the TDR surveillance testing requirements to avoid testing at power without significantly reducing the assurance of system integrity implied by the Technical Specifications. The NRC Headquarters staff agreed to review the request on an expedited basis. This document is the result of that review.

Based on these understandings, the amendment request, dated December 21, 1988, was submitted and in this letter, the licensee requested that the amendment be treated as an emergency because, unless approved, the plant would have to be shutdown monthly in order to perform certain Channel Functional Testing (CFT). Technical Specification Tables 3.3.3-1, 3.3.3-2 and 4.3.3-1 include discrepancies that imply testing capabilities not inherent in the approved design. Strict interpretation of these Tables impose monthly CFT on the 3-second time delay relay and require shutdown to implement this testing. The next CFT for Division 1 and 2 will be January 8, 1989. Unless approval of the requested change is granted before January 8, shutdown will be required to implement this testing.

Table 4.3.3-1 would be revised to note that the secondary TDR is tested during Logic System Functional Testing and exempted from the monthly CFT. Further the HPCS (Division 3) design is not described accurately by Table 3.3.3-1 channel descriptions nor are testing capabilities reflected accurately in Table 4.3.3-1. Absent the clarification provided in the changes requested by the licensee, the monthly CFT required by Table 4.3.3-1 for Division 3 would require plant shutdown for testing by January 9, 1989.

These proposed changes would provide for implementation of the HPCS degraded voltage design objective but would not increase significantly the probability of an accident caused by failure of the relay to perform its intended function. The proposed changes would preclude the impending shutdown. The NRC staff believes the licensee has not abused the emergency provisions in this instance. Accordingly, the Commission has determined that there are emergency circumstances warranting prompt approval by the Commission.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(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 or safety.

This amendment has been evaluated against the standards in 10 CFR 50.92. A discussion of these standards as they relate to the amendment request follows:

Standard 1 - Involve a significant increase in the probability or consequences of an accident previously evaluated.

The SE evaluated the design objectives of the degraded voltage instrumentation including the control logic. No design or operational change is authorized by this amendment. The worst case failure conservatively resulting in the loss of a 4.16 KV bus is an analyzed event. The consequence of this loss is not affected by the testing schedule. The safety evaluation concluded that the revised testing frequency adequately ensures operability of the degraded voltage protection instrumentation. The increase in the probability of effects equivalent to the loss of the bus due to failure of the degraded voltage protection instrumentation because of a reduced testing schedule is insignificant. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Standard 2 - Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not vary, affect or provide any physical changes to the facility. This proposed change only affects the schedule for testing existing instrumentation. Since the instrumentation is unchanged and since the reliability of the instrumentation is adequately ensured by the revised testing schedule, this amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3 - Involve a significant reduction in a margin of safety.

The requested amendment does not involve a significant reduction in a margin of safety. Since these instruments experience very little drift, the change in testing frequently would not adversely affect instrument reliability. The instrumentation remains as previously reviewed. The amendment avoids unnecessary shutdown for testing.

The staff, therefore, has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration.

Accordingly, the Commission has determined that this amendment involves no significant hazards considerations.

5.0 CONTACT WITH STATE OFFICIAL

The NRC staff advised the Washington Energy Facility Siting Council of the final determination of no significant hazards consideration by telephone on January 5, 1989. The State had no comment on this determination.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation and use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration finding with respect to this amendment. 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.

7.0 CONCLUSION

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

Principal Contributors: R. Samworth and J. Bradfute

Dated: January 6, 1989