Docket No. 50-397

May 27, 1994

Mr. J. V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations
Washington Public Power Supply System
P. O. Box 968
Richland, Washington 99352-0968

Dear Mr. Parrish:

SUBJECT: ISSUANCE OF AMENDMENT FOR THE WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2 (TAC NO. M88839)

The Commission has issued the enclosed Amendment No. 122 to the Facility Operating License No. NPF-21 for WPPSS Nuclear Project No. 2. The amendment changes the Technical Specifications (TS) in response to your February 17, 1994, application and May 13, 1994, clarification letter.

The amendment affects your 10-year hydrostatic testing requirements. The changes would:

- add a special test exception for inservice leak testing and hydrostatic testing
- add a new minimum reactor vessel metal pressure-temperature curve for ≤ eight effective full power years
- delete Table B 3/4.4.6-1 *Reactor Vessel Toughness* from the TS bases

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:

L. Mark Padovan, Acting Project Manager
Project Directorate IV-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.122 to NPF-21
2. Safety Evaluation

cc w/enclosures:
See next page

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JRoe	JClifford	GHill (2)	ACRS (10)	Region IV (4)

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OFFICE	PDIV-3/LA	PDIV-3/APM	PDIV-3/PM	OGC	PDIV-3/D
NAME	DFoster-Curseen	MPadovan	~~JClifford~~	MZOBLER	TQuay BEH for
DATE	5/24/94	5/26/94	~~/ /94~~	5/27/94	5/27/94

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DATE	5/24/94	5/26/94	~~/ /94~~	5/27/94	5/27/94

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Mr. J. V. Parrish
Washington Public Power Supply System

WPPSS Nuclear Project No. 2
(WNP-2)

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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. 122
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 February 17, 1994, supplemented by letter dated May 13, 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. 122 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

B E H for

Theodore R. Quay, Director
Project Directorate IV-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 27, 1994

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 122 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
xx(a)	xx(a)
xxiv	xxiv
1-10	1-10
3/4 4-18	3/4 4-18
--------	3/4 4-21b
--------	3/4 10-7
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	--------
B 3/4 10-1	B 3/4 10-1

LIST OF FIGURES

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
P	Prior to each radioactive release.
N.A.	Not applicable.

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown# ***	> 200°F****
4. COLD SHUTDOWN	Shutdown# ## ***	≤ 200°F****
5. REFUELING*	Shutdown or Refuel** #	≤ 140°F

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exceptions 3.10.1 and 3.10.3.

***The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled provided that the one-rod-out interlock is OPERABLE.

****See Special Test Exception 3.10.7.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for $\bar{E}$ Determination	At least once per 6 months*	1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the primary coolant system is restored to within its limits.

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1 or 3.4.6.1.c* (1) curve A or A' for hydrostatic or leak testing; (2) curve B or B' for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 80°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown, and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1 or 3.4.6.1.c curves A, A', B, B', or C, as applicable, at least once per 30 minutes.

---

*Figure 3.4.6.1.c A' and B' curves are effective for less than or equal to 8 EFPY of operation.

WNP-2 PRESSURE/TEMPERATURE LIMITS FOR 8 EFPY

TESTING AND NONNUCLEAR HEATING CURVES A'& B'

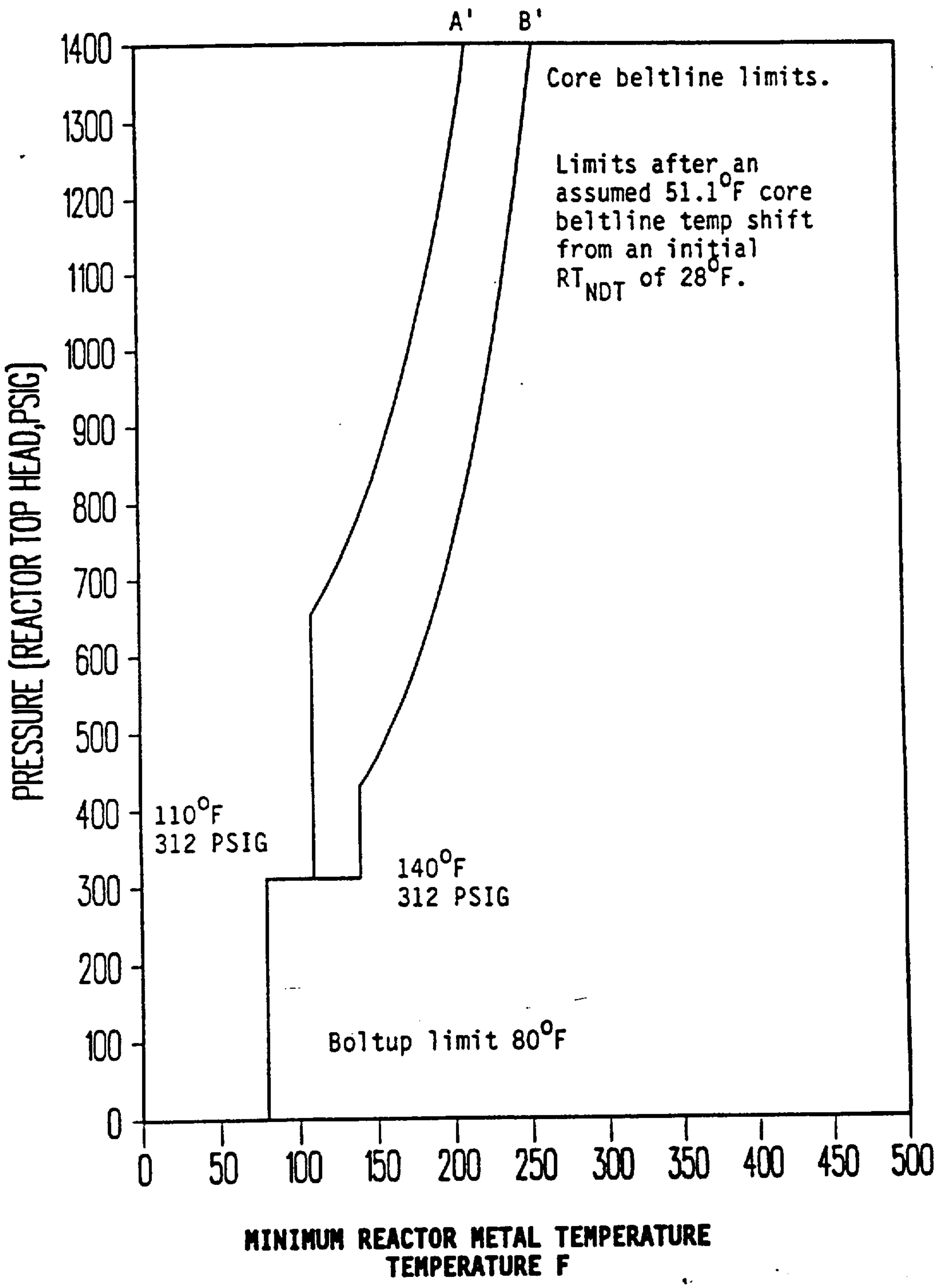

FIGURE 3.4.6.1.c

3/4.10.7 INSERVICE LEAK AND HYDROSTATIC TESTING

LIMITING CONDITION FOR OPERATION

3.10.7 When conducting Reactor Vessel inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased above 200°F, and operation considered not to be in OPERATIONAL CONDITION 3, to allow performance of an in service leak or hydrostatic test provided the maximum reactor coolant temperature does not exceed 212°F and the following OPERATIONAL CONDITION 3 LCO's are met:

a. LCO 3.1.3.8, "Control Rod Drive Housing Support";

b. LCO 3.3.2, "Isolation Actuation Instrumentation," Items 2a, 2c, and 2d of Table 3.3.2-1;

c. LCO 3.6.5.1, "Secondary Containment Integrity";

d. LCO 3.6.5.2, "Secondary Containment Automatic Isolation Valves";

e. LCO 3.6.5.3, "Standby Gas Treatment"; and

f. LCO 3.8.4.3, "Motor-Operated Valves Thermal Overload Protection."

APPLICABILITY: OPERATIONAL CONDITION 4 with average reactor coolant temperature >200°F and ≤212°F

ACTION:

With the requirements of the above specification not satisfied, immediately enter the applicable condition of the affected specification or immediately suspend activities that could increase the average reactor coolant temperature or pressure and reduce the average reactor coolant temperature to ≤ 200°F within 24 hours.

SURVEILLANCE REQUIREMENTS

4.10.7 Verify applicable OPERATIONAL CONDITION 3 surveillances for specifications listed in 3.10.7 are met.

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2-hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady-state operation will not exceed small fractions of the dose guidelines of 10 CFR Part 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron irradiation, E greater than 1 MeV, will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content, and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curves, Figure 3.4.6.1 and 3.4.6.1.c include predicted adjustments for this shift in RT_{NDT} for the end of life fluence and are effective for 10 EFPY and 8 EFPY, respectively.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1 and 3.4.6.1.c shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1 and 3.4.6.1.c for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Access to permit inservice inspections of components of the reactor coolant system is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

COMPONENT	MATERIAL TYPE	CU %	Ni %	HIGHEST STARTING RT_{NDT} °F	50 FT-LB/35 MIL TEMP°F LONG	50 FT-LB/35 MIL TEMP°F TRANS	MAXIMUM ΔRT_{NDT}* °F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS
BELTLINE									
Ring 1 Plate	SA-533, GRB, CL1	0.15	0.6	-10	+28		41	>100	
Ring 2 Plate	SA-533, GRB, CL1	0.15	0.5	-30	-8		33	>100	
Girthweld	E8018NM	0.03	1.01	N.A.	-50		36		
Girthweld	RAC01NMM	0.08	0.8	N.A.	-44		15		
NON-BELTLINE									
Ring 3 Plate	SA-533, GRB, CL1								
Ring 4 Plate	SA-533, GRB, CL1								
Vessel Flange	SA-508, CL2								
Top Head Flange	SA-508, CL2								
Top Head Dollar Plate	SA-533, GRB, CL1								
Top Head Side Plates	SA-533, GRB, CL1								
Bottom Head Dollar Plates	SA-533, GRB, CL1								
Bottom Head Radial Plates	SA-533, GRB, CL1								
Nozzles	SA-508, CL2								
Flange Bolt Studs	SA-540, B23								

*Regulatory Guide 1.99, Revision 2, calculated ΔRT_{NDT}

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirements ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access the startup and test program could be restricted and delayed.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

3/4.10.7 INSERVICE LEAK AND HYDROSTATIC TESTING OPERATION

This special test exception allows reactor vessel inservice leak and hydrostatic testing to be performed in OPERATIONAL CONDITION 4 with the maximum reactor coolant temperature not exceeding 212°F. The additionally imposed OPERATIONAL CONDITION 3 requirement for secondary containment operability provides conservatism in the response of the unit to an operational event. This allows flexibility since temperatures of the reactor vessel metal will be ≥ 180°F during the testing and a higher reactor coolant temperature will be necessary to sustain the vessel metal temperature. The flexibility is provided so that there is margin to allow temperature drift due to decay and mechanical heat.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 122 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

Washington Public Power Supply System (WPPSS) submitted a February 17, 1994, letter to the NRC requesting changes to the Technical Specifications (TS) for Nuclear Project No. 2. Their May 13, 1994, letter supplemented this information. The proposed changes would:

- add special test exception TS 3.7.10 that applies during inservice leak testing and hydrostatic testing
- add a new minimum reactor vessel metal pressure-temperature curve for $\leq$ eight effective full power years (EFPY)
- delete Table B 3/4.4.6-1 *Reactor Vessel Toughness* from the TS bases

Discussions of these items follow.

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC) Section XI requires the licensee to perform certain reactor coolant system (RCS) inservice hydrostatic and system leakage pressure tests once every 10 years. Normally, the licensee must do these tests with the average reactor coolant temperature > 200°F. This puts the plant into Operational Condition 3 (hot shutdown). However, WPPSS wants to be able to consider the plant to be in Operational Condition 4 (cold shutdown) during the testing, while keeping average reactor coolant temperature > 200°F but not > 212°F. Considering the plant to be in Operational Condition 4 allows WPPSS to:

- do outage-related maintenance on certain emergency core cooling systems (ECCS) that are not required to be operable
- not have primary containment established during testing

Doing outage-related maintenance on these systems during RCS hydrostatic and system leakage pressure testing reduces the refueling outage duration. Not having primary containment integrity during testing allows plant personnel to inspect the RCS.

The licensee's proposed special test exception would allow WPPSS to—

- consider the plant to be in Operational Condition 4 while keeping average reactor coolant temperature > 200°F but not > 212°F
- suspend TS 3.4.9.2 Operational Condition 3 residual heat removal shutdown cooling mode requirements
- require implementing certain other Operational Condition 3 requirements, including maintaining secondary containment and the standby gas treatment system operable.

The problem in keeping the plant in Operational Condition 4 during testing is that TS 3.4.6.1 gives RCS pressure and temperature (P-T) limits for the tests. Current TS 32-EFPY P-T curves require reactor vessel metal temperature to be about 236°F for hydrostatic testing. This exceeds the ≤ 200°F average reactor coolant temperature TS limit for plant Operational Condition 4. Thus, WPPSS has proposed using an 8-EFPY P-T curve where metal temperature will have to be only 180°F for testing.

Additionally, WPPSS proposes to delete Table B 3/4.4.6-1 *Reactor Vessel Toughness* from the TS basis. This table contains specific reactor vessel material composition design information which already is in the WNP-2 Final Safety Analysis Report (FSAR). WPPSS indicates that the information in the TS basis

- does not clarify the P-T curves
- makes the bases more complicated and harder to use
- can be removed from the TS bases without affecting TS adequacy

2.0 EVALUATION

We evaluated the licensee's request to (1) add a special test exception, (2) add a ≤ 8-EFPY P-T curve, and (3) delete Table B 3/4.4.6-1 from the TS bases below.

2.1 Special Test Exception TS 3.10.7

Various TS limiting conditions for operation (LCOs) apply under Operational Condition 3 and Operational Condition 4. Additional TS apply when transitioning from Mode 4 to Mode 3. This assures that the plant will have adequate shutdown cooling capability, containment integrity, and reactivity control.

Operating in Mode 4 during hydrostatic or leak testing with reactor coolant temperature above 200°F is an exception to Mode 3 requirements. Significant exceptions are not having primary containment operable, and not having the full complement of redundant ECCS. However, secondary containment remains operable under the test exception when primary containment is open for RCS inspection.

WNP-2 FSAR Section 15.6.4 describes postulated main steam line break (MSLB) accident analysis. MSLB outside containment accident analysis conservatively bounds the consequences of a RCS leak under pressure testing conditions with secondary containment integrity maintained. The FSAR analysis assumes the following:

- the reactor is operating at a power level which will cause the maximum primary system mass release

- RCS pressure is 1060 psig (higher than the 1005 → 1035 psig test pressure range the licensee indicated in their May 13, 1994, letter)

- an instantaneous circumferential MSLB occurs

Comparing these assumptions to the expected test conditions shows that a RCS leak during testing will not challenge secondary containment as severely as a MSLB. The following three factors contribute to this conclusion:

- reactor heat and recirculation pump energy input to the coolant are well below the reactor power assumed in MSLB analysis

- a leak would rapidly depressurize the RCS due to the solid plant condition

- frequent RCS leak inspections during testing should readily detect small failures before they develop further

The resulting exposures determined in the FSAR analysis are a fraction of 10 CFR 100 offsite dose requirements. Proposed TS 3.10.7 also requires that the following equipment remain operable:

- secondary containment automatic isolation valves (TS 3.6.5.2)

- standby gas treatment system (TS 3.6.5.3)

- associated automatic actuation instrumentation (TS 3.3.2)

The FSAR analysis and TS required operable equipment assure that any potential airborne contamination from RCS leaks will be within 10 CFR 100 limits.

The special test exception allows suspending TS 3.5.1 requirements that the three ECCS divisions be operable. TS 3.5.2, "ECCS - Shutdown" will then be in force, requiring two of the five ECCS systems to be operable. The licensee considers this advantageous since it would allow ECCS maintenance during the testing, thereby shortening the outage duration.

The reactor vessel would rapidly depressurize if a large RCS leak occurred. This allows the low pressure core cooling systems to operate. The TS 3.5.2 required low pressure coolant injection and core spray subsystems are adequate to maintain the core covered and prevent fuel damage for any mode 4 condition, regardless of the decay heat level. Using the test exception will allow coolant temperature to be 12°F higher than allowed in Mode 4. This is not a large enough temperature difference to alter the ability of the TS 3.5.2 required ECCS to successfully respond to a LOCA when the utility uses the test exception. The low reactor decay heat conditions expected after refueling outages (approximately forty days following shut down) will add assurance that the required ECCS capabilities will successfully counter a LOCA.

TS 3.10.7 also requires that the control rod drive housing supports are in place when the utility uses the test exception. TS 3.1.3.8 requires this in Modes 1, 2 and 3. Having the supports in place is a prudent measure to assure that a control rod will not withdraw if a control rod drive housing fails when the licensee pressurizes the reactor vessel.

The proposed Special Test Exception permits suspending TS 3.4.9.2 required shutdown cooling capability. We do not find this to be acceptable. TS 3.4.9.2 requires two operable residual heat removal (RHR) loops in Mode 4. TS 3.4.9.2 also currently requires shutdown cooling to be in operation unless a recirculation pump is operating. However, it specifically allows the utility to stop shutdown cooling during hydrostatic testing.

The licensee explains that they need to suspend TS 3.4.9.2 so they can stop shutdown cooling operation and do maintenance on the RHR system. TS 3.4.9.2, as written, and the testing conditions that maintain recirculation pumps running allow stopping shutdown cooling operation during the testing. The TS 3.4.9.2 Action Statement requires the licensee to use alternate shutdown cooling methods if a RHR shutdown cooling mode loop becomes inoperable. This requirement emphasizes the importance of maintaining shutdown cooling capability for response to potential loss of cooling events. The TS assures that operators will not eliminate cooling capabilities during unusual shutdown plant configurations. Suspending all TS 3.4.9.2 provisions to maintain operability of the shutdown cooling mode of RHR or an alternative means of cooling is not necessary. It could also potentially lead to shutdown cooling unavailability and is thus not acceptable. The remaining part of TS 3.10.7 is acceptable with the phrase "and the requirements of LCO 3.4.9.2, 'Reactor Coolant System - Cold Shutdown,' may be suspended" removed from the proposed TS.

The NRC's Acting Project Manager talked with WNP's Manager of Regulatory Programs on May 16, 1994, about this issue. The licensee acknowledged that TS 3.10.7 should not suspend LCO 3.4.9.2 requirements, and understood that the NRC will remove WNP's proposed phrase "and the requirements of LCO 3.4.9.2, 'Reactor Coolant System - Cold Shutdown,' may be suspended" from proposed TS 3.10.7. The licensee will document this understanding in a letter to the NRC.

2.2 New 8-Effective Full Power Year Pressure-Temperature Curve

We used the following NRC regulations and guidance to evaluate the P-T limits:

- 10 CFR Part 50, Appendix G
- Generic Letter 88-11
- Regulatory Guide (RG) 1.99, Rev. 2
- Standard Review Plan (SRP) Section 5.3.2

Appendix G of 10 CFR Part 50 requires that "...pressure-temperature limits for the reactor vessel must be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code...." Appendix G also puts requirements on the minimum temperature for criticality, the closure head flange, and hydrostatic pressure tests or leak tests.

Generic Letter 88-11 requires licensees to use RG 1.99, Rev. 2 methods to predict neutron irradiation effects on reactor vessel materials. This guide defines the adjusted reference temperature (ART) to be the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method. SRP 5.3.2 describes a step-by-step P-T limits calculation using fracture mechanics methodology that Appendix G to the ASME Code, Section III specifies.

The proposed P-T limits were based on the limiting material (plate C1272-1) adjusted reference temperature. The plate contains 0.15% copper and 0.6% nickel. The initial RT_{ndt} is 28°F. The licensee calculated a reference temperature shift of 51.1°F at the 1/4T (T is the beltline vessel thickness plus cladding) location based on a fluence of 1.7E17 neutron/cm^2 for the limiting plate. We identified the same plate as the limiting material, and we verified that the licensee's calculated temperature shift of 51.1°F is acceptable. The adjusted reference temperature is 79.1°F (51.1°F plus the initial RT_{ndt} of 28°F). We substituted the ART of 79.1°F into equations in SRP 5.3.2, and verified that the proposed P-T limits for hydrotest and non-nuclear heating meet Appendix G of the ASME Code.

Appendix G of 10 CFR Part 50 also imposes pressure and temperature requirements based on the closure head flange reference temperature. Appendix G, paragraph IV.A.2, has special requirements when the primary system pressure exceeds 20 percent of the preservice hydrostatic test pressure. In this case, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 requires the minimum permissible temperature to be 60°F above the closure flange reference temperature when water level is within the normal range for power operation and pressure is less than 20 percent of the hydrotest pressure. We determined that the proposed P-T limits include this requirement, based on the licensee's reported 20°F flange nil-ductility transition reference temperature.

2.3 Deleting Table B 3/4.4.6-1 From the Technical Specifications Bases

Table B 3/4.4.6-1 *Reactor Vessel Toughness* contains specific reactor vessel material composition design information. The following WNP-2 FSAR tables give similar, more detailed information:

- 5.3-1a
- 5.3-1b
- 5.3-2
- 5.3-3
- 5.3-4
- 5.3-5
- 5.3-6
- 5.3-7
- 5.3-8

WPPSS indicates the information does not clarify the P-T curves, and makes the TS bases more complicated and harder to use. There does not appear to be any benefit from maintaining this information in the TS bases. We thus agree that the utility can remove the information from the TS bases without affecting TS adequacy.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 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 14902). 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

We find that the proposed Special Test Exception, TS 3.10.7, as modified to remove the phrase "and the requirements of LCO 3.4.9.2, 'Reactor Coolant System - Cold Shutdown,' may be suspended" is acceptable. We also find that the proposed hydrotest and non-nuclear heating P-T limits conform to 10 CFR Part 50, Appendix G requirements and Generic Letter 88-11. Accordingly, the licensee may incorporate the limits into the plant TS. The limits are valid for less than or equal to 8 EFPY. The licensee may also delete Table B 3/4.4.6-1 *Reactor Vessel Toughness* from the TS bases.

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.

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