



UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

June 15, 1994

Docket No. 50-397

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 AND EXEMPTION FOR THE WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2 (TAC NOS. M88448 AND M89123)

The Commission has issued the enclosed Amendment No. 124 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 December 20, 1993, application supplemented March 25, 1994, and April 25, 1994.

You will be replacing some resiliently-seated containment vent and purge valves with new metal-to-metal seated valves during the 1994 refueling outage. The amendment changes the containment purge and vent valve TS as follows:

- removes the requirement limiting valve open position to less than or equal to 70°
- removes the surveillance requirement verifying the valves are blocked from opening more than 70° when operating the valves
- makes administrative changes to delete an out-of-date note, relocates an action statement requirement from the TS surveillance section to the TS action statement section, and changes a related TS reference to this surveillance section

There is no need for the TS to limit valve opening position to less than or equal to 70°. The resiliently-seated valves have a permanently installed mechanical stop to limit the open position to assure adequate closure times. The metal-to-metal seated valves are designed to close from the 90° open position. The TS surveillance requirement to verify the block each time the valves are used is also thus unnecessary.

The Commission is also granting the enclosed exemption regarding a requirement in Appendix J, III.D.3 of 10 CFR Part 50. The exemption extends an existing April 29, 1987, exemption to the new metal-to-metal seat containment purge supply and exhaust valves you are installing. The exemption allows you to perform Type C leak testing of the metal-to-metal seated valves at a 27-month maximum interval. We find that granting the exemption is authorized by law, will not present an undue risk to public health and safety, is consistent with the common defense and security, and meets the special circumstances described in 10 CFR 50.12(a)(2)(ii).

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June 15, 1994

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. The exemption is being forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by:
Theodore R. Quay for

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

Enclosures:

1. Amendment No. 124 to NPF-21
2. Safety Evaluation
3. Exemption

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*See previous concurrence.

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DATE	6/8/94	/ /94	5/16/94	6/7/94	6/15/94	6/15/94

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Sincerely,

Original signed by:
Theodore R. Quay for

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

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*See previous concurrence.

OFFICE	PDIV-2/LA <i>dic</i>	PDIV-2/PM	OGC	PDIV-2/D	DRPW/AD <i>AD</i>	DRPW/D <i>DR</i>
NAME	DFoster-Curseen	MPadovan	EHoller*	TQuay <i>me</i>	EAdensam	JRoe
DATE	6/8/94	1/94	5/16/94	6/7/94	6/15/94	6/15/94

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Sincerely,



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

Enclosures:

1. Amendment No. 124 to NPF-21
2. Safety Evaluation
3. Exemption

cc w/enclosures:
See next page

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Washington Public Power Supply System

WPPSS Nuclear Project No. 2
(WNP-2)

cc:

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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. 124
License No. NPF-21

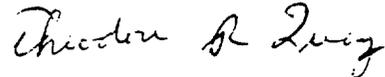
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by the Washington Public Power Supply System (licensee) dated December 20, 1993, as supplemented March 25 and April 25, 1994, complies with the standards and requirements of the Atomic Energy Act 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. 124 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 15 days from date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Theodore R. Quay, Director
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 15, 1994

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 124 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

3/4 6-4
3/4 6-11
3/4 6-12
B 3/4 6-2
B 3/4 6-3

INSERT

3/4 6-4
3/4 6-11
3/4 6-12
B 3/4 6-2
B 3/4 6-3

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- d. Type B and C tests shall be conducted with gas at P_a , 34.7 psig,* at intervals no greater than 24*** months except for tests involving:
 - 1. Air Locks
 - 2. Main steam line isolation valves,
 - 3. Valves pressurized with fluid from a seal system,
 - 4. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and
 - 5. Purge supply and exhaust isolation valves with resilient seals.
- e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.
- f. Main steam line isolation valves shall be leak tested at least once per 18 months.
- g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least $1.10 P_a$, 38.2 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.
- h. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.
- i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.1 and 4.6.1.8.2.
- j. The provisions of Specification 4.0.2 are not applicable to 24-month or 40 ± 10 -month surveillance intervals.

*Unless a hydrostatic test is required per Table 3.6.3-1.

***For those tests conducted during refueling outages, the 24-month interval may be exceeded by no more than 3 months.

CONTAINMENT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

- c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line isolation valve, and
- d. The combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

- a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a , 34.7 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.
- b. If any periodic Type A test fails to meet $0.75 L_a$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet $0.75 L_a$, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet $0.75 L_a$, at which time the above test schedule may be resumed.
- c. The accuracy of each Type A test shall be verified by a supplemental test which:
 1. Confirms the accuracy of the test by verifying that the supplemental test result, L_c , minus the sum of the Type A and the superimposed leak, L_o , are equal to or less than $0.25 L_a$.
 2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.
 3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be between $0.75 L_a$ and $1.25 L_a$.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.8.2 The cumulative time that the drywell and suppression chamber purge system has been in operation PURGING through the Standby Gas Treatment System shall be verified to be less than or equal to 90 hours per 365 days prior to use in this mode of operation.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and suppression chamber purge system may be in operation with the drywell and/or suppression chamber purge supply and exhaust butterfly isolation valves open for inerting, deinerting, or pressure control. PURGING through the Standby Gas Treatment System shall be restricted to less than or equal to 90 hours per 365 days.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With a drywell and/or suppression chamber purge supply and/or exhaust butterfly isolation valve open for other than inerting, deinerting, or pressure control, close the butterfly valve(s) within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With a drywell and suppression chamber purge supply and/or exhaust isolation valve(s) with resilient material seals having a measured leakage rate exceeding $0.05 L_a$ per valve test, and the leakage added to the previously determined total for all valves and penetrations subject to Type B and C tests per LCO 3/4.6.1.2 is less than $0.6 L_a$, secure the valves in the closed position and perform maintenance at the next plant cold shutdown to reduce leakage to within 4.6.1.8.1.a, otherwise, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8.1 At least once per 6 months, on a STAGGERED TEST BASIS, each 24- and 30-inch drywell and suppression chamber purge supply and exhaust isolation valve with resilient material shall be demonstrated OPERABLE by verifying that the measured leakage is:

- a. Less than or equal to $0.05 L_a$ per valve test or,
- b. Greater than $0.05 L_a$ per valve test, the leakage added to the previously determined total for all valves and penetrations subject to Type B and C tests per LCO 3/4.6.1.2 shall be less than $0.6 L_a$,
- c. In the event the valves are to be operated, and 4.6.1.8.1.a. has been exceeded, a leakage test must be performed within 24 hours following operation, to ensure compliance with $0.6 L_a$.

CONTAINMENT SYSTEMS

BASES

MSIV LEAKAGE CONTROL SYSTEM (Continued)

Design specifications require the system to accommodate a leak rate of five times the Technical Specification leakage allowed for the MSIVs while maintaining a negative pressure downstream of the MSIVs. The allowed leakage value per each valve is 11.5 scfm, or a total of 230 scfh (3.8 scfm).^(a) When corrected for worst case pressure, temperature and humidity expected to be seen during surveillance testing conditions, the flow would never exceed an indicated value (uncorrected reading from local flow instrumentation) of 5 cfm. The 30 cfm acceptance criterion provides significant margin to this design basis requirement and provides a benchmark for evaluating long term blower performance. The Technical Specification limit for pressure of -17" H₂O W.C. was also established based on a benchmark of the installed system performance capability. This -17" H₂O W.C. provides assurance that the negative pressure criterion can be met.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 34.7 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 34.7 psig does not exceed the design pressure of 45 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 2 psid. The limit of 1.75 psig for initial positive containment pressure will limit the total pressure to 34.7 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The 24-inch and 30-inch drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, de-inerting and pressure control. Until all the drywell and suppression chamber valves have been qualified as capable of closing within the times assumed in the safety analysis, they shall not be open more than 90 hours in any consecutive 365 days. Valves not capable of closing from a full open position during a LOCA or steam line break accident shall be blocked so as not to open more than 70°.

(a) Letter, G02-75-238, dated August 18, 1975, NO Strand (SS) to OD Parr (NRC), "Response to Request for Information Main Steam Isolation Valve Leakage Control System"

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 34.7 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

CONTAINMENT SYSTEMS

BASES

DEPRESSURIZATION SYSTEMS (Continued)

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus, there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below 200°F during any period of relief valve operation with sonic conditions at the discharge exit for quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety/relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety/relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety/relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures for those isolation valves designed to close automatically that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

CONTAINMENT SYSTEMS

BASES

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

The time limit on use of the drywell and suppression chamber purge lines is not restricted when using the 2-inch purge supply and exhaust isolation valves since the 2-inch valves will close during a LOCA or steam line break accident and therefore the SITE BOUNDARY dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during PURGING operations. The design of the 2-inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations."

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. Valves with metal to metal seals will be tested on a Type C schedule in accordance with Surveillance 4.6.1.2.d to assure allowable leakage rates are not exceeded. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of those valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 34.7 psig which is below the design pressure of 45 psig. Maximum water volume of 128,827 ft³ results in a downcomer submergence of 12 ft and the minimum volume of 127,197 ft³ results in a submergence approximately 4 inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 124 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 20, 1993, supplemented with March 25, 1994, and April 25, 1994, letters, the Washington Public Power Supply System (the licensee) requested an amendment to the WPPSS Nuclear Project No. 2 (WNP-2) Technical Specifications (TS). The proposed amendment would revise limiting conditions for operations (LCOs) and surveillance requirements (SRs) relating to containment vent and purge system isolation valves. The additional information contained in the supplemental letters was clarifying in nature, was within the scope of the initial notice, and did not affect the NRC staff's proposed no significant hazards consideration determination.

2.0 BACKGROUND

The WNP-2 facility TS presently include requirements relating to (a) blocking devices intended to limit the maximum opening angle of containment vent and purge valves, (b) augmented testing requirements for resiliently-seated containment vent and purge valves, and (c) cumulative time limits on containment venting and purging during normal operation.

The WNP-2 containment vent and purge systems contain large butterfly-type isolation valves. These valves are normally closed during power operation. The valves are opened intermittently to adjust containment pressure or inert or de-inert the containment with nitrogen while starting up or shutting down the reactor. Thus, the valves must be capable of quick closure against the high pressure and dynamic forces resulting from a loss of coolant accident (LOCA). SRP Section 6.2.4/BTP CSB 6-4 specifies a maximum closure time of five seconds for such valves. In order to meet these performance requirements, containment vent and purge isolation valves typically have a mechanical stop device that prevents them from being capable of opening more than a predetermined amount, typically 60° to 70° (90° is full open). The available actuator torque is capable of closing the valve within the time limit, against the dynamic forces calculated to result from a LOCA, with the valve position limited to no more than the calculated amount. Utilities typically use add-on auxiliary devices capable of being readjusted or removed by operating personnel to limit containment isolation valve travel. A surveillance requirement is included in the TS to verify that a butterfly valve opening angle does not exceed its limit since add-on devices can be

removed or misadjusted. This surveillance must be performed each time a valve is opened during power operation, unless it has been performed within the previous 31 days.

3.0 EVALUATION

Presently installed WNP-2 containment vent and purge butterfly isolation valves are limited to a maximum opening angle of 70°. WNP-2 does not utilize travel-limiting stops that are subject to removal or misadjustment. Instead, the stops are as welded extensions in the actuators. The licensee plans to replace two vent and two purge valves with metal-to-metal seated valves. These valves will be able to close within the required time limit, against LOCA dynamic forces, from the full 90°-open position. Thus, the licensee requested that the TS requirements relating to containment vent and purge valve opening angle limits be deleted. The staff considered the licensee's request, and determined that it is acceptable. This conclusion is based on the physical impossibility that the existing valves will be opened in excess of 70° during plant operation, and the fact that the future replacement valves need not be limited to less than 90° open.

Leakage tests of large butterfly valves installed in containment vent and purge systems have shown that resilient valve seat materials in these valves have a history of relatively rapid and severe failures (Reference: I&E Circular 77-11 dated September 6, 1977). Because of this, the TS include augmented testing requirements to ensure that seal degradation is detected and repaired in a timely manner. Large resiliently-seated vent and purge valves are required to be leak tested at 6-month intervals. In addition, the TS specify a leakage limit of 0.05 L_a for each valve. These requirements exceed those of 10 CFR Part 50, Appendix J, which specifies a 24-month test interval with no individual valve leakage limits, only a combined total leakage limit of 0.6 L_a for all penetrations.

The licensee requested changes to the TS to allow the new metal-to-metal seated valves to be tested in accordance with the 24-month test interval specified in 10 CFR Part 50, Appendix J, as modified by an exemption granted by our April 29, 1987, letter. The exemption authorizes alternate testing criteria to accommodate the unique WNP-2 annual, Spring refueling schedule. It allows up to 27 months between local leak rate tests in lieu of 24 months as prescribed by Appendix J, but imposes more stringent leak rate test acceptance criteria. Because the new valves do not rely on resilient seating seals, inclusion of these valves in the Type C testing interval of 24 months specified in 10 CFR Part 50, Appendix J is acceptable. The licensee's request to extend the exemption for a 27-month test interval is addressed in the separate exemption.

The licensee also requested deleting an out-of-date TS footnote pertaining to the containment purge cumulative time limit. When venting the primary containment, it is preferable to vent via the Standby Gas Treatment System (SGTS) to provide a filtered path for release for the effluent. However, should a LOCA occur while venting the containment using this pathway, the SGTS train in use could be damaged due to overpressurization. For this reason, in

a previous amendment, the staff granted a temporary, one-time increase in the 90 hour/year limit. This was implemented by the addition of a note to SR 4.6.1.8.3. The note expired April 10, 1988. Since the note is no longer effective, the licensee has proposed deleting it. The staff finds deletion of this note 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 14901). 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.

Pursuant to 10 CFR 51.32, an environmental assessment of the exemption from certain requirements of 10 CFR Part 50 Appendix J, related to these actions was published in the Federal Register on May 25, 1994 (59 FR 27075). Accordingly, the Commission has determined that the issuance of this exemption will not result in any environmental impacts beyond those evaluated in WNP-2's Final Environmental Statement.

6.0 CONCLUSION

The staff has reviewed the licensee's application to amend TS relating to containment purge and vent valves. The staff has performed a safety review of the supporting information provided by the licensee, and has evaluated the proposed changes against standard acceptance criteria for TS. Based on this review, the staff has determined that the proposed changes are acceptable.

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: W. O. Long

Date: June 15, 1994

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)	
Washington Public Power Supply System)	Docket No. 50-397
Nuclear Project No. 2 (WNP-2))	
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)	

EXEMPTION

I.

Washington Public Power Supply System (the licensee) is the holder of Facility Operating License No. NPF-21 which authorizes operation of the WNP-2 Nuclear Plant at steady-state reactor power levels not in excess of 3323 megawatts thermal. The WNP-2 facility is a boiling water reactor located on Hanford Reservation in Benton County near Richland, Washington. The license provides, among other things, that it is subject to all rules, regulations, and Orders of the Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

II.

Paragraph III.D.3 of Appendix J to 10 CFR Part 50 requires, in part, that "Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than two years." By letter dated April 29, 1987, the staff issued an exemption from the requirement for Type C testing during each reactor refueling shutdown, and an extension of the maximum interval from 24 months to 27 months for Type B and C testing. This

exemption specifically excluded Containment Purge Supply and Exhaust Valves, which the staff required to continue to be tested at the existing 6-month interval.

Pursuant to 10 CFR 50.12(a), the NRC may grant exemptions from the requirements of the regulations (1) which are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and (2) where special circumstances are present.

III.

By letter dated March 25, 1994, the licensee requested an exemption from 10 CFR 50 Appendix J to allow Type C testing of Containment Purge Supply and Exhaust Valves with metal-to-metal seats at 27-month intervals. The licensee plans to replace approximately half of the valves during the 1994 refueling outage, and the remainder at a future date. No change to the 6-month test interval is requested for the remaining Containment Purge Supply and Exhaust Valves that have resilient seats. In a letter dated December 20, 1993, regarding an associated technical specification change request, the licensee stated that the new valves have been proven by industry experience and design to be capable of maintaining design requirements for leakage over an extended period of time. The licensee indicated that the replacement valves will be required to meet even tighter permissible leakage limits. Extending the maximum allowable interval between tests to 27 months is requested to allow for variations in the weather-related length of the approximately annual operating cycle from year to year. Details concerning the justification for extending the Type C test interval from 24 to 27 months are contained in the staff's letter granting the exemption dated April 29, 1987.

IV.

Accordingly, the Commission concluded that the licensee's proposed test schedule for the metal-to-metal seated Containment Purge Supply and Exhaust Valves is acceptable, and can be tested at a 27-month maximum interval. The remaining valves with resilient seats will continue to be tested every 6 months.

The special circumstances for granting this exemption pursuant to 10 CFR 50.12 have also been identified. As stated in part in 10 CFR 50.12(a)(2)(ii), special circumstances are present when application of the regulation in the particular circumstance is not necessary to achieve the underlying purpose of the rule. Application of the resilient-seated valve leak test requirements to metallic-seated valves would increase surveillance and maintenance costs for no increased safety benefit. The vendor certifies that appropriate leakage criteria are met, as applicable. The licensee states that the valve design, specifications, and qualification documentation for these valves verify that Type C leakage testing intervals are appropriate. The special circumstance of 10 CFR 50.12(a)(2)(ii) for extending the Type C leakage test interval from 24 months to 27 months is as described in the staff's letter granting the exemption dated April 29, 1987. Consequently, the Commission concludes that the special circumstances of 10 CFR 50.12(a)(2)(ii) exist in that application of the regulation in these particular circumstances is not necessary to achieve the underlying purpose of the rule.

V.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, this exemption as described in Section III. above is authorized by law, will not present an undue risk to the public health and safety, and is

consistent with the common defense and security. The Commission further determines that special circumstances as provided in 10 CFR 50.12(a)(2)(ii) are present justifying the exemption.

Therefore, the Commission hereby grants an exemption from the requirement for Type C testing during each reactor refueling shutdown, with an extension of the maximum interval from 24 months to 27 months for Type C testing, as described in Section III. above.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will have no significant impact on the environment (59 FR 27075).

This exemption is effective upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Elinor G. Adensam, Acting Director
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland
this 15th day of June 1994