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

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

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.

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 pentrate 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 $0.25 \cdot L_a$.
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.

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.

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.

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" H_2O W.C. was also established based on a benchmark of the installed system performance capability. This -17" H_2O 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.

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.

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 ft^3 results in a downcomer submergence of 12 ft and the minimum volume of 127,197 ft^3 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.