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. 128
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 6, 1993, as supplemented by letter dated May 6, 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. 128 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

James W. Clifford for

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: July 8, 1994

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 128 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 3-74	3/4 3-74
3/4 4-7a	3/4 4-7a
3/4 5-5	3/4 5-5
B 3/4 4-1a	B 3/4 4-1a

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 - With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. In lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 - With the number of OPERABLE Safety/Relief Valve Position Indicator instrumentation channels less than the Minimum Channels OPERABLE requirement of Table 3.3.7.5-1,

a. Restore an inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours, and

b. Verify operability and perform daily surveillance of the Tailpipe Temperature Monitoring instrument for the affected SRV until the Minimum Channels OPERABLE requirement is satisfied. Absent an OPERABLE Tailpipe Temperature monitor for the affected SRV restore the inoperable Tailpipe Temperature Monitor to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R	1,2
2. Reactor Vessel Water Level	M	R	1,2
3. Suppression Chamber Water Level	M	R	1,2
4. Suppression Chamber Water Temperature	M	R	1,2
5. Suppression Chamber Air Temperature	M	R	1,2
6. Primary Containment Pressure	M	R	1,2
7. Drywell Air Temperature	M	R	1,2
8. Drywell Oxygen Concentration	M	R	1,2
9. Drywell Hydrogen Concentration	M	R	1,2
10. Safety/Relief Valve Position Indicators*	M	R#	1,2
11. Suppression Chamber Pressure	M	R	1,2
12. Condensate Storage Tank Level	M	R	1,2
13. Main Steam Line Isolation Valve Leakage Control System Pressure	M	R	1,2
14. Neutron Flux:			
APRM	M	R	1,2
IRM	M	R	1,2
SRM	M	R	1,2
15. RCIC Flow	M	R	1,2
16. HPCS Flow	M	R	1,2
17. LPCS Flow	M	R	1,2

*This includes acoustic monitor, valve stem position, and tailpipe temperature instrument channels.

#The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow are adequate to perform the test.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 a) The safety valve function of at least 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 safety/relief valves @ 1150 psig +1%/-3%
4 safety/relief valves @ 1175 psig +1%/-3%
4 safety/relief valves @ 1185 psig +1%/-3%
4 safety/relief valves @ 1195 psig +1%/-3%
4 safety/relief valves @ 1205 psig +1%/-3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

b) The safety valve function of at least 4 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 safety/relief valves @ 1150 psig +1%/-3%
4 safety/relief valves @ 1175 psig +1%/-3%
4 safety/relief valves @ 1185 psig +1%/-3%
4 safety/relief valves @ 1195 psig +1%/-3%
4 safety/relief valves @ 1205 psig +1%/-3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3, when THERMAL POWER is less than 25% of RATED THERMAL POWER.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 90°F, close the stuck open safety/relief valve(s); if unable to close the open

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With both the acoustic monitor and valve stem position indicator for one or more safety/relief valve(s) inoperable, restore either the acoustic monitor or valve stem position indicator to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 The position indicators for each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months.**

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow are adequate to perform the test.

SURVEILLANCE REQUIREMENTS (Continued)

e. For the ADS by:

1. At least once per 31 days by verifying that the accumulator backup compressed gas system pressure in each bottle is ≥ 2200 psig.

2. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas system low pressure alarm system.

3. At least once per 18 months:

a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

b) Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig* and observing that either:

1) The control valve or bypass valve position responds accordingly, or

2) There is a corresponding change in the measured steam flow.

c) Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system low pressure alarm system and verifying an initiation setpoint of ≥ 140 psig on decreasing pressure and an alarm setpoint ≥ 135 psig on decreasing pressure.

d) Verifying the nitrogen capacity in at least two accumulator bottles per division within the backup compressed gas system.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow are adequate to perform the test.

3/4 5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

a. The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

e. The high pressure core spray (HPCS) system with a flow path capable of taking suction from one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

   1. From the suppression chamber, or

   2. When the suppression pool level is less than the limit or is drained, from the condensate storage tank containing at least 135,000 available gallons of water, equivalent to a level of 13.25 feet in a single condensate storage tank or 7.6 feet in each condensate storage tank.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5*.

ACTION:

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

b. With both of the above required subsystems/systems inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/system to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

3/4.4.2 SAFETY/RELIEF VALVES (Continued)

the dual purpose safety/relief valves in their ASME Code qualified mode (spring lift) of safety operation.

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients that represent the most severe abnormal operational transient resulting in a nuclear system pressure rise. The evaluation of these events with the final plant configuration has shown that the MSIV closure is slightly more severe when credit is taken only for indirect derived scrams; i.e., a flux scram. Utilizing this worse case transient as the design basis event, a minimum of 12 safety/relief valves are required to assure peak reactor pressure remains within the Code limit of 110% of design pressure.

Testing of safety/relief valves is normally performed at lower power with adequate steam pressure and flow. It is desirable to allow an increased number of valves to be out of service during testing. Therefore, an evaluation of the MSIV closure without direct scram was performed at 25% of RATED THERMAL POWER assuming only 4 safety/relief valves were operable. The results of this evaluation demonstrate that any 4 safety/relief valves have sufficient flow capacity to assure that the peak reactor pressure remains well below the code limit of 110% of design pressure.

TMI Action Plan Item II.D.3, "Direct Indication of Relief and Safety Valve Position," states that reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe. Each WNP-2 SRV has both a valve stem position indication device and an acoustic monitor flow detection device which independently meet the requirements of Item II.D.3. Hence failure of one device does not impact compliance to II.D.3 and entry into Limiting Condition for Operation action statement 3.4.2.c is required only for inoperability of both devices associated with a specific SRV.

Demonstration of the safety/relief valve lift settings will be performed in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

The primary containment sump flow monitoring system monitors the UNIDENTIFIED LEAKAGE collected in the floor drain sump with a sensitivity such that 1 gpm change within 1 hour can be measured. Alternatively, other methods for measuring flow to the sump which are capable of detecting a change in UNIDENTIFIED LEAKAGE of 1 gpm within 1 hour with an accuracy of ± 2% may be used, for up to 30 days, when the installed system is INOPERABLE.

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and been found to be acceptable provided the unit is operated in accordance with the single recirculation loop operation Technical Specifications herein.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. Where the recirculation loop flow mismatch limits cannot be maintained during two recirculation loop operation, continued operation is permitted in the single recirculation loop operation mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve capacity is designed to limit the primary system pressure, including transients, in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1971, Nuclear Power Plant components (up to and including Summer 1971 Addenda). The Code allows a peak pressure of 110% of design pressure (1250 (design) X 1.10 = 1375 psig maximum) under upset conditions. In addition, the Code specifications require that the lowest valve setpoint be at or below design pressure and the highest valve setpoint be set so that total accumulated pressure does not exceed 110% of the design pressure.

The safety valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by one of two sources; i.e., a direct position switch or neutron flux signal. The direct scram signal is derived from position switches mounted on the main steamline isolation valves (MSIV's) or the turbine stop valve, or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing, and following 10% travel of full stroke. The pressure switches are actuated when a fast closure of the control valves is initiated. Further, no credit is taken for power operation of the pressure relieving devices. Credit is only taken for