May 23, 2000

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: WNP-2 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION 3.4.9, RESIDUAL HEAT REMOVAL SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN (TAC NO. MA6166)

Dear Mr. Parrish:

The Commission has issued the enclosed Amendment No. 164 to Facility Operating License No. NPF-21 for WNP-2. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated July 29, 1999, as supplemented by letter dated January 31, 2000.

The amendment revises the applicability of TS 3.4.9 from "Mode 3 with steam dome pressure less than the RHR cut in permissive" to "Mode 3 with steam dome pressure less than 48 psig." Notes associated with TS Surveillance Requirements 3.4.9.1 and 3.5.1.2 and the associated Bases are also changed to reflect the 48 psig limit.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
/RA/

Jack Cushing, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. Amendment No. 164 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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

ENERGY NORTHWEST

DOCKET NO. 50-397

WNP-2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 164
License No. NPF-21

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Energy Northwest dated July 29, 1999, as supplemented by letter dated January 31, 2000, 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. 164and 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. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 23, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 164

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Appendix A Technical Specifications with the attached revised 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
3.4.9-1	3.4.9-1
3.4.9-3	3.4.9-3
3.5.1-4	3.5.1-4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System — Hot Shutdown

LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

---------------------------NOTES---------------------------

1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period.

2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances.

APPLICABILITY: MODE 3 with reactor steam dome pressure less than 48 psig.

ACTIONS

---------------------------NOTES---------------------------

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1	Initiate action to restore RHR shutdown cooling subsystem to OPERABLE status.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour
	AND		
	A.3	Be in MODE 4.	24 hours
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1	Initiate action to restore one RHR shutdown cooling subsystem or one recirculation pump to operation.	Immediately
	AND		
	B.2	Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND		
	B.3	Monitor reactor coolant temperature and pressure.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	------------------NOTE------------------ Not required to be met until 2 hours after reactor steam dome pressure is less than 48 psig. -- Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time of Condition E or F not met. <u>OR</u> Two or more required ADS valves inoperable.	G.1 Be in MODE 3. <u>AND</u> G.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
H. HPCS and Low Pressure Core Spray (LPCS) Systems inoperable. <u>OR</u> Three or more ECCS injection/spray subsystems inoperable. <u>OR</u> HPCS System and one or more required ADS valves inoperable. <u>OR</u> Two or more ECCS injection/spray subsystems and one or more required ADS valves inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.1.2	-------------------NOTE------------------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than 48 psig in MODE 3, if capable of being manually realigned and not otherwise inoperable. --- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.1.3	Verify ADS accumulator backup compressed gas system average pressure in the required bottles is $\geq$ 2200 psig.	31 days

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

WNP-2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated July 29, 1999, the Washington Public Power Supply System (now Energy Northwest), the licensee for WNP-2, proposed changes to Technical Specification (TS) 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown." The proposed TS changes include changing the Limiting Condition for Operation (LCO) Applicability statement for TS 3.4.9 from "MODE 3 with reactor steam dome pressure less than the RHR cut-in permissive pressure," to "MODE 3 with reactor steam dome pressure less than 48 psig." Changes to notes associated with TS Surveillance Requirements (SRs) 3.4.9.1 and SR 3.5.1.2 and the associated Bases are also needed to clarify that parts of the RHR system are not analyzed to operate at the temperature associated with the RHR cut-in permissive pressure. The staff requested additional information by letter of January 3, 2000, and the licensee provided a response dated January 31, 2000.

The supplemental letter dated January 31, 2000, provided clarifying information, did not expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration determination published in the Federal Register on August 25, 1999 (64 FR 46430).

2.0 EVALUATION

Energy Northwest requested a change to the WNP-2 technical specifications (TSs) in accordance with 10 CFR Parts 50.59, 50.90, and 2.101. This TS change request is submitted consistent with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," December 29, 1998. The proposed revision of TS 3.4.9 and changes to the notes associated with TS Surveillance Requirements (SRs) 3.4.9.1 and SR 3.5.1.2 and associated Bases are discussed below.

TS 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," establishes requirements for fuel cooling during hot shutdown conditions, when the irradiated fuel generates heat during the decay of fission products and increases the temperature of the reactor coolant. The decay heat must be removed to reduce the temperature of the reactor coolant to less than or equal to 200°F in preparation for cold shutdown maintenance operations or core refueling, or the decay heat must be removed for maintaining the reactor in the hot shutdown condition.

The licensee has requested TS changes to include changing the applicability for TS 3.4.9 from "MODE 3 with reactor steam dome pressure less than the RHR cut-in permissive pressure," to "MODE 3 with reactor steam dome pressure less than 48 psig." The proposed changes involve TS 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," and changes to the notes associated with TS SRs 3.4.9.1 and SR 3.5.1.2, and associated Bases.

This change is needed because some parts of the RHR system downstream of the RHR heat exchangers are not analyzed to operate at the temperature associated with the RHR cut-in permissive pressure. The cut-in permissive pressure has an allowable value of less than or equal to 135 psig, which causes an isolation of the RHR Shutdown Cooling (SDC) Mode, to prevent an intersystem loss-of-coolant accident (LOCA) during a shutdown by preventing the premature initiation of RHR and to also isolate the RHR if reactor pressure exceeds 135 psig. The proposed revision does not change the protection against an intersystem LOCA provided by the pressure switches. This pressure interlock is provided only for equipment protection to prevent an intersystem LOCA scenario and credit for the interlock is not assumed in the accident or transient analysis in the final safety analysis report (FSAR). This change does propose to continue to use administrative controls to protect portions of the RHR system against unanalyzed thermal stress that could occur if the RHR SDC is manually initiated at or above 48 psig under saturated conditions.

The SDC mode of the RHR system is not initiated at the cut-in permissive pressure because of original design temperature limitations of 335°F, corresponding to a saturation pressure of 95 psig. A further restriction of 295°F, corresponding to a saturation pressure of 48 psig, was put in place downstream of the RHR heat exchangers, where lower temperatures were expected in the SDC mode for the downstream piping and pipe supports.

During a conference call with the staff on November 17, 1999, the licensee indicated that a Non-Conformance Report (NCR) in 1988 had documented this potential discrepancy and had initiated an evaluation and corrective actions. Subsequent to the conference call, the staff transmitted a request for additional information (RAI) to the licensee dated January 3, 2000, requesting details of the evaluation. The RAI specifically asked for the effects on thermal stress and fatigue cycle of the affected piping system due to potential higher operating temperatures. The licensee responded by letter dated January 31, 2000.

The licensee stated that NCR 288-028, in February 1988, noted that the RHR piping downstream of the heat exchanger was designed for a normal operating temperature of 295 ^{0}F, while by existing plant procedure it was possible to expose a portion of that piping to a higher temperature during shutdown. The licensee also stated that, in 1988, an evaluation was performed to assess the condition of the RHR SDC piping system. The licensee further stated that since the time of NCR 288-028, plant procedures were changed to limit RHR SDC operation to the 295°F limit.

The licensee stated that the affected RHR SDC supply and return piping consists of a combination of American Society of Mechanical Engineers (ASME) Code Class 1 and Code Class 2 piping. The licensee also indicated that, based on the plant operating history, the plant had been started up 34 times by the end of 1988. Although every shutdown did not include going into the shutdown cooling mode, the licensee conservatively assumed that 34 higher temperature cycles had been experienced. The licensee further stated that, based on its RHR

Class 1 and Class 2 piping stress analyses, the resulting piping stresses met the respective ASME Code Class 1 and 2 allowable stress limits. The licensee also indicated that, in 1988, several piping supports were inspected and no damage was found. The staff finds that the licensee's analyses and evaluation are acceptable. With respect to thermal fatigue cycle limits, the licensee stated that an evaluation accounting for the increase in temperature for initiation of RHR SDC was performed. The results demonstrated that the cumulative usage factor is within the ASME Class 1 piping fatigue limit of 1.0. The results also demonstrated that for the Class 2 piping, the resulting thermal fatigue cycle is well within the 7000-cycle limit. In addition, the licensee stated that the occurrence of higher temperature RHR SDC was noted in the applicable system design calculations and will be accounted for in any future updates of the ASME Class 1 fatigue analyses or evaluations for plant life evaluation. The staff finds the licensee's evaluation acceptable.

The staff finds that the proposed TS change is based on the original plant design operating temperature for the RHR SDC piping. It will provide assurance that the temperature limits of the piping supports will not be exceeded and is, therefore, acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

TS 3.4.9

The proposed TS change is that the applicability statement be changed from "MODE 3 with reactor steam dome pressure less than the RHR cut-in permissive pressure," to "MODE 3 with reactor steam dome pressure less than 48 psig." This is more restrictive in that the cut-in permissive pressure is 135 psig.

TS SR 3.4.9.1 and SR 3.5.1.2

These SR notes are changed to be consistent with the change from " ---less than the RHR cut-in permissive pressure." to "--- less than 48 psig."

The associated Bases 3.4.9 and 3.4.10 are also consistently modified to clarify the change to 48 psig and 295°F bases.

Based on the staff's review of the proposed TS changes and the licensee's responses to the staff RAI for the thermal limit analyses, the staff finds the proposed TS changes and SR notes acceptable because they are more restrictive than the current TS pressure limit, provide additional protection to the RHR piping system in the SDC mode, and are consistent with the operational limits required by the FSAR and current plant procedures.

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 (64 FR 46430). 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.

6.0 CONCLUSION

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 Contributors: Edward Kendrick
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Date: May 23, 2000