WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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October 3, 1995
GO2-95-204

Docket No. 50-397

Mr. L. J. Callan
NRC Regional Administrator
U.S. NRC, Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21 FACILITY OPERATION AT REACTOR CORE POWER LEVELS IN EXCESS OF FULL POWER**

Reference: Letter GO2-95-188, dated September 20, 1995, JV Parrish (SS) to LJ Callan (NRC), "Facility Operation at Reactor Core Power Levels in Excess of Full Power"

This follows up the referenced notification regarding WNP-2 exceeding the reactor power license limit. This letter is submitted in accordance with license condition 2.F which requires a written follow-up report within 14 days of operation in excess of full power (3486 megawatts thermal). This condition was identified on September 19, 1995 when the preliminary results of the Reactor Feedwater (RFW) flow tests performed by General Electric (GE) indicated 2.46% greater feedwater flow than was indicated by plant instrumentation. Since this indicated that actual reactor power was greater than licensed full power, WNP-2 reactor power was lowered to less than 97.5% of indicated rated power. The plant computer was adjusted to provide RFW flow and reactor power consistent with the preliminary data, as a new 100% power indication.

On September 25, 1995, GE provided preliminary results from a second RFW flow test which apparently confirm the results from the initial test. The second test indicated 2.13% greater feedwater flow than was indicated by plant instrumentation prior to September 19, 1995. Both recent tests have used Rubidium. The previous test, performed in 1990, used a sodium tracer and verified the accuracy of installed instrumentation.

The Supply System has evaluated various causes for the potentially anomalous RFW flow indications. The plant computer flow calculation components were verified unchanged since the sodium tracer test.

The RFW flow instrumentation is believed to be correct (to within at least 1.3%) but inconsistent with the Rubidium test results. The Supply System is therefore postulating a change in the relationship between RFW flow and differential pressure in the flow venturis. Potential causes still under evaluation include venturi throat erosion, cracks in the venturi welds, or a buildup of material which alters the dynamics at the pressure taps. Also under investigation is the possibility that the Rubidium and/or sodium injection tests were inaccurate.

Additional Planned Actions:

1. The preliminary results from the two flow tests are expected to be finalized October 6, 1995.

2. The Supply System is considering conduct of a sodium tracer test to gather additional data.

3. The results of the additional flow test may establish calibration and inspection requirements for the flow venturis. If required, the visual inspections will be performed during R-11.

4. Upon completion of the flow tests and identification of discharge coefficients, the plant computer system tolerances will be reviewed and adjusted as necessary. This will be performed on completion of all preliminary and planned testing and is presently intended to be completed by November 30, 1995.

5. Calculations for determining the past safety significance of a potential 2.5% reactor power error over the last five years continue. Preliminary analysis completed to date indicates no safety significance. The calculation effort is expected to be completed by November 30, 1995.

6. On verification of the 2.5% error, vessel fluence and the potential effect on reactor vessel pressure and temperature limit curves will be evaluated. This effort is expected to be completed by December 30, 1995.

7. The effect on reactor internal consumable items such as control rod blades, local power range monitors (LPRMs), and isotopic inventory of discharged fuel and fuel in the reactor will be evaluated. This effort is expected to be completed by December 30, 1995.

The investigation performed to date verifies that no actual reactor safety issue exists.

Should you have any questions or desire additional information, please call D. A. Swank at (509) 377-4563.

Sincerely,

J. V. Parrish (Mail Drop 1023)
Vice President, Nuclear Operations

JVP/LCF/ml

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