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Director
Office of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Additional Information Concerning RCC
Guide Thimble Plug Removal

This letter and the attached documents are being submitted in response to NRC Staff comments from Mr L Phillips and Mr H Balukjun concerning our April 22, 1985 submittal, entitled "Safety Evaluation for RCC Guide Thimble Plug Removal."

Attachment A contains revised pages to our safety evaluation (NSPNAD-8412) submitted with the letter referenced above. Contained in the revised pages for Section 5.1 is a discussion of the liftoff forces for both Exxon and Westinghouse fuel assemblies. The removal of thimble plugs will make the core less resistive to flow. The resulting flow increase is accompanied by a decrease in pump differential pressure which is characteristic of centrifugal pumps. Since the resistance of the rest of the system, i.e. piping and steam generators, remain the same, the pressure drop across the rest of the system will increase with the increasing flow. The differential pressure across the pump equals the sum of the pressure drop across the rest of the system plus the pressure drop across the core. With the pump differential pressure decreasing and the pressure drop across the rest of the system increasing, it is clear that the pressure drop across the core will decrease. Therefore, the removal of the thimble plugs will drop the differential pressure across the core and reduce the liftoff forces acting on the fuel, decreasing the possibility of fuel liftoff.

Please contact us if you have questions concerning this issue.

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DMM/TMP/tp

Attachment A, Revised pages for NSPNAD-8412

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
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ATTACHMENT

NORTHERN STATES POWER COMPANY

Revised Pages to NSPNAD-8412

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3.6 Safety Limit Curves

The safety limit curves for Prairie Island, given in Technical Specifications Section 2.1, along with the associated overpower and overtemperature ΔT setpoints were calculated using $F\Delta H=1.55$ and $FQ=2.51$ (Reference 5). Tech Spec Figure 2.1-1 shows the currently accepted safety limit curves.These are calculated as the loci of points where MDNBR=1.3. A spectrum of bounding points have been renalyzed using the accepted NSPNAD methods (NSPNAD-8102P Rev.2) assuming $F\Delta H=1.55$, $FQ=2.32$ and $W_{BYP}=6.0\%$. Table 3 shows these results. Since all points are greater than 1.3, the current safety curves (T.S. Figure 2.1-1) bound operation with the thimble plugs removed. These values bound PI 1 Cycles 9 and 10 and PI 2 Cycles 9 and 10.

TABLE 3.3

Power (% Rated)	Pressure (psia)	Average Temp (°F)	MDNBR
92	2400	611	1.492
100	2400	605	1.470
120	2400	584	1.447
100	2250	598	1.455
120	2250	576	1.425
59	1700	593	2.255
100	1700	573	1.449
120	1700	550	1.361

4.3 LOCA-ECCS Analysis

LOCA-ECCS analyses for both fuel types, i.e. Exxon TOPROD and Westinghouse OFA, are currently being performed with 6.0% bypass flow. These analyses will be in place for the Prairie Island Unit 1 Cycle 11 startup. The thimble plugs will remain in the core until this time.

4.4 ATWS Analysis

The current analysis (ref: Prairie Island Updated Safety Analysis Report) is based on similar, but not necessarily bounding, plant parameters.

The purpose of this analysis is to show that if the Prairie Island Units were modified with the Alternate Mitigating Systems Actuation Circuitry (AMSAC), it would satisfactorily meet the proposed alternative 3 as defined in Volume 3 of NUREG 0460, which requires "modifications to reduce susceptibility to common mode electrical failures and to provide mitigation of most ATWS events." This issue is currently being reviewed by the NRC.

Therefore no reanalysis or Technical Specification changes are required for removal of the thimble plugs since there is no licensing requirement, at this time, to provide a plant specific bounding ATWS analysis.

5.0 MECHANICAL AND STRUCTURAL ANALYSIS

The fuel assembly mechanical design and the vessel internals structural design assume upper limit bounding flows. Removal of the thimble plugs decreases the active core flow and increases the total vessel flow due to the decreased bypass flow resistance.

5.1 Fuel Mechanical Design and Fuel Liftoff Forces Analysis

Exxon and Westinghouse have calculated the liftoff flow for their fuels to be 206,000 gpm or greater (Reference 11 and 12) for the Prairie Island plants.

The current best estimate RCS flow rates for Prairie Island Units 1 and 2 are 196,883 and 195,331 gpm respectively. These flows will increase 0.6% due to removal of tne thimble plugs (Section 2.2) and approximately 0.6% due to the lower resistance of the Westinghouse fuel (based on a total Westinghouse core). Also a 2.0% measurement and instrument uncertainty must be applied (Table 5.1). The revised flow rates are then 203,237 and 201,635 gpm for Prairie Island Units 1 and 2 respectively. Since these flow rates are bounded by the above analyses, the thimble plugs can safely be removed.

5.2 Reactor Vessel and Internals Mechanical and Vibration Analysis

A review of the Westinghouse reactor vessel and internals mechanical and vibration design analysis shows that thimble plugs are not a consideration. In addition, original hot functional tests at Prairie Island were run at 140% rated flow (249,200 gpm) and showed no adverse effects. Therefore the design analysis will remain bounding after the thimble plugs are removed.

TABLE 5.1

RCS FLOW RATE

Measurement & Instrument Uncertainties

Instrument

• Leading edge flow meter	
LEFM absolute accuracy and repeatability	± .67%
• Elbow Taps	
Absolute accuracy (from LEFM)	± .67%
Repeatability	± .15%
ΔP transmitter accuracy	± .6%
DVM Calibration Accuracy	± .6%
DVM Absolute Accuracy	± .6%
RMS Total Accuracy	± 1.94%*

* 2.0% will be used

9.0 References

1. "Prairie Island Nuclear Generating Plant Units 1 & 2, Updated Safety Analysis Report", Docket Numbers 50-282, 50-306.

2. "Reload Safety Evaluation Methods for Application to PI Units" NSPNAD-8102P, December 1982.

3. XN-NF-80-61, "Prairie Island Nuclear Plants TOPROD Safety Analysis Report", Revision 1, March 1981.

4. XN-75-32(P)(A), Supplements 1, 2, 3, 4, "Computational Procedure for Evaluating Fuel Rod Bowing" October 1983.

5. WCAP 8090, "Fuel Densification Prairie Island Nuclear Generating Plant Unit No. 1," March 1973.

6. "Prairie Island Units 1 Cycle 10 Final Reload Design Report (RSE)" NSPNAD-8411P, October 1984.

7. "Prairie Island Unit 2 Cycle 9 Final Reload Design Report (RSE)" NSPNAD-8404P Rev.2, May 1984.

8. XN-NF-83-38, "Prairie Island Units 1 and 2 Limiting Break LOCA/ECCS Analysis using EXEM/PWR" May 1983.

9. WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis.

10. "Flow of Fluids Through Valves, Fittings, and Pipe," Crane, Technical Paper No. 410.

11. Telecopy NLG:020:85, N.Garner (Exxon) to T.M.Parker (NSP) "Evaluation of Increased Coolant Flow at Prairie Island" August 26, 1985.

12. Letter 85NS-G-023, R.T.Meyer (Westinghouse) to T.M.Parker (NSP) "Release of Westinghouse Proprietary Information to NRC" August 22, 1985.