Wisconsin Electric POWER COMPANY

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VPNPD-96-095

10 CFR 50.4
10 CFR 50.90

November 13, 1996

Document Control Desk
US NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555-0001

Gentlemen:

DOCKETS 50-266 AND 50-301
SUPPLEMENT TO TECHNICAL SPECIFICATIONS CHANGE REQUESTS 188 AND 189
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

This letter provides additional information in support of Technical Specifications Change Requests (TSCRs) 188 and 189. TSCRs 188 and 189 were submitted in letters dated June 4, 1996. Supplements to TSCRs have been submitted in letters dated August 5, 1996, September 26, 1996 and October 21, 1996. These requests propose amendments to the Point Beach Technical Specifications that were identified by analyses performed in support of Unit 2 operations following replacement of steam generators this fall.

We are providing additional information regarding the evaluation of the small break loss of coolant accident. We have determined that the additional information does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of any effluent release, or result in any significant increase in individual or cumulative occupational exposure. Therefore, we conclude that the proposed amendments meet the requirements of 10CFR51.22(c)(9) and that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared. The original "No Significant Hazards" determinations for operation under the proposed Technical Specifications remain applicable.

If you require additional information, please contact us.

Sincerely,

G M Krieser for

Bob Link
Vice President
Nuclear Power

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PDR ADOCK 05000266
P PDR

CAC

cc: NRC Resident Inspector
NRC Regional Administrator
PSCW

Subscribed and sworn before me on
this 13th day of November, 1996.

Jacqueline Gourk

Notary Public, State of Wisconsin
My commission expires 10/26/2000

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Additional Information for Technical Specifications Change Requests 188 and 189
Small Break LOCA Evaluation

In a letter dated August 5, 1996, the following information was provided in support of the safety evaluation of the proposed Technical Specification change to incorporate the average reactor coolant system temperature range of 557 to 573.9°F:

> <u>Small Break LOCA (FSAR §14.3.1)</u>
>
> A sensitivity analysis performed for another two-loop Westinghouse PWR shows that a lower full power average reactor coolant system temperature results in lower PCT for small break loss-of-coolant accidents. Therefore, no PCT assessment is necessary to account for the proposed full power average RCS temperature range.

During the review of this information, the NRC reviewer questioned that the small break LOCA analysis was performed with an assumed average reactor coolant system temperature of 570°F and that the safety evaluation only discussed the effects of lower average reactor coolant system temperatures. Wisconsin Electric provided information in a letter dated October 21, 1996, in response to this question. That response stated that although the analysis documented in the Point Beach FSAR (§14.3.1) shows that a 570°F average reactor coolant was assumed as the initial condition for the small break LOCA, the accident analysis was based on an average reactor coolant system temperature of 573.9°F. The 3.9°F difference in reactor coolant system average temperature was based on the Westinghouse analysis that generates LOCA specific parameters for the small break LOCA analysis.

It was subsequently decided that the effect on Peak Clad Temperature (PCT) for average reactor coolant system temperatures higher than 570°F could be evaluated for the small break LOCA analysis. The proposed reactor coolant system average temperature limit of 573.9°F has been evaluated based on sensitivity studies previously submitted to the NRC for the Rochester Gas and Electric R. E. Ginna Nuclear Plant (Reference 1). These analyses were reviewed by the NRC as documented in Reference 2. The small break LOCA analysis for the R. E. Ginna Nuclear Plant documents a sensitivity of 396°F PCT increase for each 14.5°F increase in reactor coolant system average temperature. Therefore, application of this sensitivity to the 3.9°F difference between the proposed average reactor coolant system average temperature range high limit (573.9°F) and the analysis average reactor coolant system temperature (570°F) results in a 107°F increase in PCT which will be added to the current PCT assessment for Point Beach Nuclear Plant. This results in a cumulative small break LOCA PCT assessment of 1,186°F, based on 809°F analysis result plus 270°F of previously determined margin allocation plus the 107°F margin allocation for the 3.9°F difference in reactor coolant system average temperature. The small break LOCA PCT remains well below the 10CFR50.46 limit of 2,200°F.

Although it was not specifically questioned, for conservatism an additional assessment was performed for the reactor coolant system average temperature uncertainty of +/- 4°F. Using the +4°F uncertainty would result in an additional 110°F in small break LOCA PCT, which in turn results in a small break LOCA PCT assessment of 1296°F. This is still well below the 10CFR50.46 limit of 2,200°F.

The small break LOCA analysis for Point Beach Nuclear Plant is based on NRC approved methods contained in WCAP-10054-P-A (Proprietary), WCAP-10081 (Non-Proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

The evaluation of the effect of lower reactor coolant system temperatures on small break LOCA PCT, as stated previously, remains applicable. Therefore, a total PCT margin allocation of +217°F will be applicable for the small break LOCA PCT assessment.

<u>References</u>

1. Letter from R. C. Mecredy (RG&E) to A. R. Johnson (NRC), dated June 19, 1995, "Small Break Loss of Coolant Accident Analysis."

2. Letter from A. R. Johnson (NRC) to R. C. Mecredy (RG&E), dated February 27, 1996, "R. E. Ginna Nuclear Power Plant Small-Break Loss-Of-Coolant Accident Analysis Model (TAC No. M92764), Docket No. 50-244.