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U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Gentlemen:

DOCKETS 50-266 AND 50-301
30 DAY REPORT OF ECCS EVALUATION MODEL CHANGES, 10 CFR 50.46
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

As required by Title 10 of the Code of Federal Regulations Part 50.46(a)(3)(ii), Wisconsin Electric Power Company (Licensee) is submitting this report of changes to and errors discovered in emergency core cooling system (ECCS) evaluation models for Point Beach Nuclear Plant, Units 1 and 2. This letter provides a summary of ECCS evaluation model changes and errors identified since the last report dated April 18, 1996 (Letter NPL 96-0142). Model changes include changes to the small break loss of coolant accident (LOCA) model considered to be reportable under the 30 day reporting requirement in 10 CFR 50.46(a)(3)(ii). A summary of the changes is provided below with additional details and a summary sheet of peak cladding temperature (PCT) margin in the attachments.

Small Break LOCA Evaluation Model

The effect of a full power average temperature (T_{avg}) range of 557°F to 573.9°F for Point Beach Nuclear Plant Units 1 and 2 on the small break LOCA Evaluation Model with NOTRUMP has been assessed a PCT penalty of +107°F. The penalty is based on sensitivity studies performed for another Westinghouse 2-loop plant using NOTRUMP.

An error was discovered in the SBLOCTA computer program. The error was in the adjustments made to power to compensate for the assumed pellet diameter. Adjustments were also made to the fuel rod cladding creep and strain model. The error correction results in a +10°F penalty assessed to the small break LOCA Evaluation Model with NOTRUMP.

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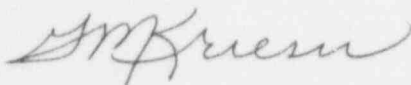
Large Break LOCA Evaluation Model

No changes have been made to and no errors have been discovered in the large break LOCA evaluation model since the last report.

The current cumulative change in PCT for small break LOCA analysis is 377°F for a total PCT of 1186°F (Attachment 2). The cumulative change in PCT for the large break LOCA is 109°F for a total PCT of 2137°F (Attachment 3). The ECCS model will be reanalyzed as part of the fuel upgrade/power uprating analysis for both units. The expected completion date for the analysis is June, 1997.

Please contact us if you have any questions about this information.

Sincerely,



Gary M. Krieser
Manager
Industry and Regulatory Services

RJK/kmc

cc: NRC Resident Inspector
NRC Regional Administrator, Region III

ECCS EVALUATION MODEL CHANGES AND ERRORS

- **T_{avg} Increase**

Wisconsin Electric (WE) submitted Technical Specifications Change Request (TSCR) 189, VPNPD-96-036, dated June 4, 1996, to implement a range of 557°F to 573.9°F for the average reactor coolant system temperature at full power. The current small-break LOCA analysis for Point Beach Nuclear Plant assumes an average reactor coolant temperature of 570°F. During review of the change request NRC requested an evaluation of the impact of the average reactor coolant system temperatures higher than 570°F on small-break LOCA PCT.

WE submitted a supplement to TSCR 189 in letter VPNPD-96-095, dated November 13, 1996 describing the evaluation performed. The evaluation uses sensitivity studies previously submitted to the NRC for the R. E. Ginna Nuclear Plant using the 1985 Westinghouse small-break LOCA Evaluation Model with NOTRUMP. The Ginna analysis includes a sensitivity study showing a 396°F increase in PCT for a 14.5°F increase in reactor coolant system average temperature. Applying this sensitivity to the 3.9°F difference between the proposed average reactor coolant system 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.

- **SBLOCTA Fuel Rod Initialization**

An error was discovered in the SBLOCTA computer program related to adjustments which are made as part of the fuel rod initialization process which is used to obtain agreement between the SBLOCTA model and the fuel data supplied from the fuel thermal-hydraulic design calculations at full power, steady-state conditions. Specifically, an adjustment to the power, which is made to compensate for adjustments to the assumed pellet diameter was incorrect. Additionally, updates were made to the fuel rod cladding creep and strain model to correct logic errors that could occur in certain transient conditions. These model revisions had a small effect on the fuel rod initialization process, and could produce small effects during the transient. Due to the small magnitude of effects, and the interaction between the two items, they are being evaluated as a single, closely related effect.

The error affects the 1985 Westinghouse small-break LOCA Evaluation Model with NOTRUMP.

Representative plant calculations with the corrected model demonstrate that these revisions result in a predicted peak cladding temperature increase on the order of +10°F.

SMALL BREAK PEAK CLADDING TEMPERATURE MARGIN UTILIZATION

Point Beach Nuclear Plant:

A. Analysis of Record (7/88)	PCT=	809 °F
B. Prior Permanent ECCS Model Assessments	Δ PCT=	47 °F
C. 10 CFR 50.59 Safety Evaluations	Δ PCT=	320 °F
D. 1996 10 CFR 50.46 Model Assessments		
1. SBLOCTA Fuel Rod Initialization Error	Δ PCT=	10 °F
E. Temporary ECCS Model Issues (none)	Δ PCT=	0 °F
F. Other Margin Allocations (none)	Δ PCT=	0 °F

Licensing Basis PCT + Margin Allocations	PCT=	1186 °F
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LARGE BREAK PEAK CLADDING TEMPERATURE MARGIN UTILIZATION

Point Beach Nuclear Plant:

A. Analysis of Record (2/91)	PCT=	2028 °F
1. Combined SSE and LOCA Events	Δ PCT=	10 °F
B. Prior Permanent ECCS Model Assessments	Δ PCT=	14 °F
C. 10 CFR 50.59 Safety Evaluations	Δ PCT=	85 °F
D. 1996 10 CFR 50.46 Model Assessments		
1. None	Δ PCT=	0 °F
E. Temporary ECCS Model Issues (none)	Δ PCT=	0 °F
F. Other Margin Allocations (none)	Δ PCT=	0 °F

Licensing Basis PCT + Margin Allocations

PCT= 2137 °F

10 CFR 50.59 SAFETY EVALUATIONS

Point Beach Nuclear Plant:

1. Small Break ECCS Safety Evaluations

A. Loss of Auxiliary Feedwater

$\Delta PCT = 213$ °F

B. T_{avg} Increase

$\Delta PCT = 107$ °F

Total 10 CFR 50.59 Small Break Assessments

PCT = 320 °F

2. Large Break ECCS Safety Evaluations

A. Reduced T_{avg} and Reduced TDF

$\Delta PCT = 85$ °F

Total 10 CFR 50.59 Large Break Assessments

PCT = 85 °F