October 28, 1994

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Mr. Robert E. Link, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

SUBJECT: ISSUANCE OF EMERGENCY AMENDMENT NOS.156 AND 160 TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27 - POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 (TACS M90642 AND M90643)

Dear Mr. Link:

The Commission has issued the enclosed Amendment Nos. 156 and 160 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The amendments revise the Technical Specifications in response to your application dated October 20, 1994.

These amendments revise Technical Specification (TS) Section 15.3.1.G, "Operational Limitations," to reduce the reactor coolant system raw measured total flow rate and operating pressure, modify TS Section 15.2.3.1.B to increase the required reduction in the ΔT trip setpoint, and modify TS Figure 15.2.1-1 to reflect new reactor core safety limits, all for Unit 2 only. The applicable bases are also revised.

A copy of the Safety Evaluation is also enclosed. The notice of issuance and final determination of no significant hazards consideration and opportunity for hearing will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by Allen G. Hansen
Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-266
and 50-301

Enclosures:
1. Amendment No. 156 to DPR-24
2. Amendment No. 160 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

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

October 28, 1994

Mr. Robert E. Link, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

SUBJECT: ISSUANCE OF EMERGENCY AMENDMENT NOS. 156 AND 160 TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27 - POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 (TACS M90642 AND M90643)

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Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

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and 50-301

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1. Amendment No.156 to DPR-24
2. Amendment No.160 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

Mr. Robert E. Link, Vice President
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

cc:

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6612 Nuclear Road
Two Rivers, Wisconsin 54241

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 156
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated October 20, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 156 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Deputy Director
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: October 28, 1994

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 160
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated October 20, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 3.B of Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 160 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Deputy Director
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: October 28, 1994

ATTACHMENT TO LICENSE AMENDMENT NOS. 156 AND 160

TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
TS 15.2.3-2	TS 15.2.3-2
TS 15.3.1-19	TS 15.3.1-19
TS Figure 15.2.1-2	TS Figure 15.2.1-2

(3) Low pressurizer pressure - ≥1865 psig for operation at 2250 psia primary system pressure
≥1790 psig for operation at 2000 psia primary system pressure

(4) Overtemperature

$$\Delta T \left(\frac{1}{1+\tau_3 S}\right)$$

$$\leq \Delta T_o \; (K_1 - K_2(T(\frac{1}{1+\tau_4 S}) - T')(\frac{1+\tau_1 S}{1+\tau_2 S} + K_3(P-P') - f(\Delta I))$$

where

ΔT_0	=	indicated ΔT at rated power, ^oF
T	=	average temperature, ^oF
T'	≤	573.9^{o}F (Unit 1)
T'	≤	570.0^{o}F (Unit 2)
P	=	pressurizer pressure, psig
P'	=	2235 psig
K_1	≤	1.30
K_2	=	0.0200
K_3	=	0.000791
τ_1	=	25 sec
τ_2	=	3 sec
τ_3	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD
τ_4	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD

and f(ΔI) is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ within -17, +5 percent, f(ΔI) = 0.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power for Unit 1, or by an equivalent of 3.1 percent of rated power for Unit 2.

G. OPERATIONAL LIMITATIONS

The following DNB related parameters shall be maintained within the limits shown during Rated Power operation:

1. T_{avg} shall be maintained below 578°F.

2. Reactor Coolant System (RCS) pressurizer pressure shall be maintained:

   a. Unit 1: ≥2205 psig during operation at 2250 psia, or
   ≥1955 psig during operation at 2000 psia.

   b. Unit 2: ≥1955 psig during operation at 2000 psia.

3. Reactor Coolant System raw measured Total Flow Rate (See Basis).

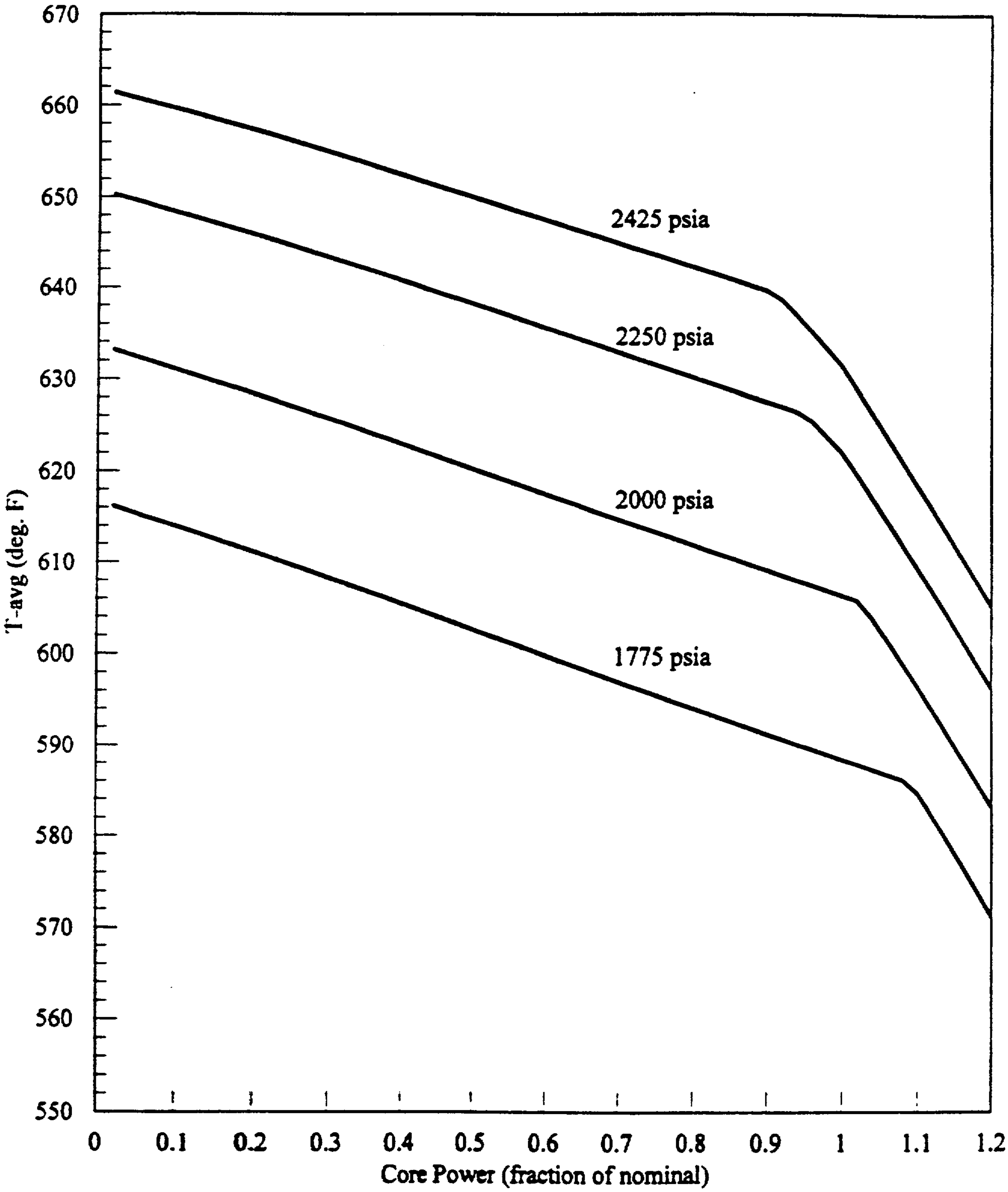
   a. Unit 1 ≥ 181,800 gpm Unit 1

   b. Unit 2 ≥ 174,000 gpm Unit 2

Basis:

The reactor coolant system total flow rate for Unit 1 of 181,800 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (178,000 gpm). The reactor coolant system total flow rate for Unit 2 of 174,000 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (170,400 gpm). The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimetric at the beginning of each cycle.

Figure 15.2.1-2
REACTOR CORE SAFETY LIMITS
POINT BEACH UNIT 2



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 156 AND 160 TO

FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter dated October 20, 1994, Wisconsin Electric Power Company (WEPCO), the licensee, pursuant to 10 CFR 50.90, requested an amendment to Facility Operating Licenses DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Units 1 and 2, respectively. The amendment proposes revisions to Technical Specifications (TSs) Section 15.3.1.G, "Operational Limitations," to reduce the reactor coolant system raw measured total flow rate and operating pressure, modify TS Section 15.2.3.1.B to increase the required reduction in the ΔT trip setpoint, and modify TS Figure 15.2.1-1 to reflect new reactor core safety limits, all for Unit 2 only, for operation through December 31, 1996. Changes were also proposed for the applicable bases. The requested changes are necessitated by steam generator (SG) tube plugging, which is currently 18.4% for SG-A and 15.6% for SG-B.

TS 15.3.1.G.3 currently specifies that the reactor coolant system (RCS) measured total flow rate must be ≥181,800 gpm for Unit 1 and ≥179,200 gpm for Unit 2. TS 15.3.1.G.2 requires that RCS pressurizer pressure be maintained at ≥2205 psig during operation at 2250 psia or ≥1955 psig for operation at 2000 psia. TS 15.2.1 states that the combination of thermal power level, coolant pressure and coolant temperature shall not exceed the limits shown on Figures 15.2.1-1 and 15.2.1-2 for Units 1 and 2, respectively. Finally, TS 15.2.3.1.B(4) specifies the instrument settings required to enforce the limits specified in Figures 15.2.1-1 and 15.2.1-2.

The proposed change to TS 15.3.1.G.3 reads as follows:

"3. Reactor Coolant System raw measured Total Flow Rate (See Basis)

a. Unit 1 ≥ 181,800 gpm Unit 1
b. Unit 2 ≥ 174,000 gpm Unit 2"

The proposed modification to TS 15.3.1.G.2 reads as follows:

"2. Reactor Coolant System (RCS) pressurizer pressure shall be maintained:

a. Unit 1: ≥2205 psig during operation at 2250 psia
or
≥1955 psig during operation at 2000 psia

b. Unit 2: ≥1955 psig during operation at 2000 psia"

The proposed change to TS 15.2.3.1.B(4)(b) reads:

"(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power for Unit 1 or by an equivalent of 3.1 percent for Unit 2."

Finally, TS Figure 15.2.1-2 and the applicable bases are modified to support the reduction of RCS flow in Unit 2.

2. TRANSIENT ANALYSIS EVALUATION

Westinghouse performed an evaluation of the proposed 5200 gpm flow reduction for Unit 2, using NRC approved methodologies. The scope of the evaluation included loss of coolant accidents (LOCA), non-LOCA transients and SG tube rupture. The 5200 gpm flow reduction is estimated to correspond to an average SG tube plugging of 24%. Based on the number of the SG tubes plugged, a flow imbalance of about 3% is anticipated. This flow imbalance has been taken into account in the analysis. In addition, the following operating parameters were specified: RCS pressure of 2000 psia, T_{avg} of 570 °F and steam generator pressure of 718 psia. The thermal design flow of 170,400 gpm was taken into account in the analysis, which results from the 174,000 gpm measured flow with a 2.1% uncertainty.

The analysis showed that peak cladding temperature increased by about 10 °F from the UFSAR analysis of record, but remains well within the 10 CFR 50.46 limits. The SG tube plugging assumed in the LOCA analysis was 20%, and an additional 5% of the tubes were assumed to be unavailable, as a result of postulated tube crush due to combined LOCA and seismic loads (Reference 1).

The result of the reduced flow on the non-LOCA transient analysis was that the thermal safety limits became more limiting at all power and pressure levels. A revised TS Figure 15.2.1-2 was derived. In addition, due to the higher temperature change across the core, an increased F(ΔI) penalty is assumed (from 2.0 to 3.1). The non-LOCA transients with non-DNB acceptance criteria were also reanalyzed and found to meet the acceptance criteria with the reduced flow.

The scope of this evaluation was to assess the potential SG tube rupture consequences, in view of the specified set of operational parameters (RCS and secondary pressures, temperatures, primary flow and percentage of tubes plugged). The estimated offsite radiation doses will increase by about 1%, but will remain well within the 10 CFR 100 limits.

LOCA, non-LOCA and SG tube rupture transient analyses were performed with NRC approved methods. The staff agrees with the licensee that the results justify the proposed TS and bases changes.

3.0 SYSTEM AND COMPONENT INTEGRITY EVALUATION

As part of the justification to support the decrease in RCS flow rate limit, the licensee's submittal evaluated the structural integrity of the reactor coolant systems and components. The evaluation assumed plant operation at an RCS pressure of 2000 psia, T_{avg} of 570°F, steam pressure of 718 psia, and reduced thermal design flow of 170,400 gpm. The evaluation was for operation through December 31, 1996.

On October 24, 1994, a conference call was held between WEPCO and NRC regarding the Westinghouse evaluation mentioned in the transmittal. WEPCO stated that the Westinghouse analysis attached to the October 19, 1993, submittal (Reference 2), was also used in the current assessment.

In October 1993, the licensee assessed continued operation of Point Beach, Unit 2, at an RCS pressure of 2250 psia, a steam pressure of 785 psia and a reduced RCS T_{avg} of 570 °F, as documented in Reference 2. The evaluation considered increased hydraulic forces, increased thermal stresses and fatigue usage on the primary loop, vessel, internals, fuels, steam generators, pressurizer and the reactor coolant pumps, as a result of increased subcooling, higher fluid densities and larger transient temperature and pressure differentials during postulated plant transients. In a safety evaluation (Reference 3), the staff concluded that continued operation of Point Beach, Unit 2 at a T_{avg} of 570 °F was acceptable through December 31, 1996, without any adverse effects on the structural integrity of the reactor coolant system and components.

The licensee evaluated the effects of a 5200 gpm reduction in RCS flow rate on the structural integrity of the reactor coolant system and components, by comparing operational conditions used in the previous analysis with the new operating conditions at a proposed thermal design flow of 170,400 gpm. Since the comparison indicated that operational conditions used in the previous evaluation are bounding for the proposed operation, the existing analyses of reactor coolant systems and components remain unchanged.

Considering the conservatism in the previous analysis, such that the actual measured fatigue cycle is about half of what was assumed in the fatigue design analyses, and the combination of stresses due to other loading conditions, such as LOCA, seismic and pressure differential, the staff concludes that operation through December 1996 at the proposed thermal design flow of 170,400 gpm has no adverse impact on the original stress and fatigue analyses of the reactor coolant system, components and their supports. This conclusion is based on plant operation within the limits specified by the licensee (including operation primarily in a base-load mode), at a nominal RCS pressure of 2000 psia, a RCS T_{avg} of 570 °F, and a steam pressure of 718 psia.

4.0 EMERGENCY CIRCUMSTANCES

Prior to the current Unit 2 refueling outage, the licensee had submitted a Technical Specification change request to allow utilization of alternative standards to evaluate the necessity to repair sleeved steam generator (S/G) tubes with flaw indications found using eddy current techniques. The staff verbally notified the licensee on October 4, 1994, that the request would be denied. Had this request been approved, the significant S/G plugging which has led to the request being evaluated herein, would not have been necessary.

Due to the staff denial of the previous request, the licensee completed the S/G inspections, and plugged 245 tubes. This resulted in a level of plugging which significantly exceeded their estimates, potentially reducing the RCS flow rate to less than the current TS limit.

Due to the possible staff denial of the licensee's request for alternative S/G tube repair standards, the licensee had initiated an evaluation to provide a basis for reduced RCS flow. Upon finally determining the need to seek an amendment to allow reduced flow, the licensee completed their evaluation in a timely manner, and forwarded it with a TS amendment request to the staff for review on October 20, 1994, with a requested approval by October 28, 1994 (the scheduled startup date for Unit 2).

The staff has concluded that an emergency situation exists in that failure to act in a timely way will prevent resumption of operation, and that the licensee could not avoid this emergency situation.

5.0 BASIS FOR FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The staff's review is presented below.

The flow reduction together with the changes in the ΔT trip setpoint automatic reduction fraction, the system operating pressure and the core safety limits, maintain adequate system operating safety margins. In addition, the new operating parameters are bounded by the existing safety analyses. Therefore, the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This change does not create the possibility of a new or different kind of accident from any accident previously evaluated, because there are no physical changes to plant systems or components, or to the way the plant is operated.

This change does not involve a significant reduction in a margin of safety, because analysis has comfirmed that the compensatory changes to offset the effects of reduced RCS flow are sufficient to ensure that safety analysis requirements are still met.

Based on this review, the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has determined that the amendment request involves no significant hazards consideration.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

These amendments change 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. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission made a final no significant hazards consideration finding with respect to this amendment. Accordingly, these amendments meet 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 these amendments.

8.0 CONCLUSION

The staff 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: L. Lois
C. Wu

Date: October 28, 1994

Attachment: Reference Sheet

REFERENCES

1. WEP-91-171, "Wisconsin Electric Power Company, Point Beach, Units 1 and 2, ECCS Evaluation Model Changes" Westinghouse Electric Corporation, June 20, 1991.

2. Wisconsin Electric Power Company letter #VPNPD-93-184, to USNRC, dated October 19, 1993.

3. USNRC letter to Wisconsin Electric Power Company, dated October 27, 1993.